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

August 7, 2008

Mr. William R. Campbell, Jr.
Chief Nuclear Officer and Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000390/2008003 AND 05000391/2008003 AND ANNUAL ASSESSMENT MEETING SUMMARY

Dear Mr. Campbell:

On June 30, 2008, 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 July 7 and August 5, 2008, with Mr. M. Skaggs and other members of your staff, and with Mr. M. Brandon, respectively.

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.

This report documents one NRC identified and one self-revealing finding of very low safety significance (Green), which involved a violation of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because they have been entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs), consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCV in this report you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the United States Nuclear Regulatory Commission, ATTN.: Document Control Desk Washington DC 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington D.C. 20555-0001; and the NRC Resident Inspector at the Watts Bar facility.

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

Sincerely,

/RA/

Eugene F. Guthrie, 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 05000390/2008003, 05000391/2008003
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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cc w/encl:
Gordon P. Arent
New Generation Licensing Manager
Tennessee Valley Authority
Electronic Mail Distribution

Masoud Bajestani
Vice President Watts Bar Unit 2
Watts Bar Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

Ashok S. Bhatnagar
Senior Vice President
Nuclear Generation Development and Construction
Tennessee Valley Authority
Electronic Mail Distribution

Michael K. Brandon
Manager
Licensing and Industry Affairs
Tennessee Valley Authority
Electronic Mail Distribution

William R. Campbell, Jr.
Chief Nuclear Officer and Executive Vice President
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

(Vacant)
Vice President
Nuclear Support
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

General Counsel
Tennessee Valley Authority
Electronic Mail Distribution

John C. Fornicola
General Manager
Nuclear Assurance
Tennessee Valley Authority
Electronic Mail Distribution

Gregory A. Boerschig
Plant Manager
Watts Bar Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

Larry E. Nicholson
General Manager
Performance Improvement
Tennessee Valley Authority
Electronic Mail Distribution

Michael A. Purcell
Senior Licensing Manager
Nuclear Power Group
Tennessee Valley Authority
Electronic Mail Distribution

Michael J. Lorek
Vice President, Nuclear Engineering and Projects
Tennessee Valley Authority
Electronic Mail Distribution

Michael D. Skaggs (WB2)
Site Vice President
Watts Bar Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

Beth A. Wetzel
Manager
Corporate Nuclear Licensing and Industry Affairs
Tennessee Valley Authority
Electronic Mail Distribution

(cc w/encl cont’d – See next page)

cc w/encl cont'd
Senior Resident Inspector
Watts Bar Nuclear Plant
U.S. Nuclear Regulatory Commission
1260 Nuclear Plant Road
Spring City, TN 37381-2000

County Executive
375 Church Street
Suite 215
Dayton, TN 37321

County Mayor
P.O. Box 156
Decatur, TN 37322

Lawrence Edward Nanney
Director
Division of Radiological Health
TN Dept. of Environment & Conservation
Electronic Mail Distribution

James H. Bassham
Director
Tennessee Emergency Management
Agency
Electronic Mail Distribution

Ann Harris
341 Swing Loop
Rockwood, TN 37854

Letter to William R. Campbell, Jr. from Eugene F. Guthrie dated August 7, 2008

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000390/2008 AND 05000391/2008

Distribution w/encl:
C. Evans, RII
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T. Sullivan, NRR

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-390, 50-391
License Nos.:	NPF-90 and Construction Permit CPPR-92
Report Nos.:	05000390/2008003, 05000391/2008003
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Units 1 and 2
Location:	Spring City, TN 37381
Dates:	April 1, 2008 - June 30, 2008
Inspectors:	R. Monk, Senior Resident Inspector M. Pribish, Resident Inspector R. Chou, Reactor Inspector (Section 4OA5.3) L. Moore, Senior Reactor Inspector (Section 4OA5.2) L. Suggs, Reactor Inspector (Section 4OA5.4)
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000390/2008-003, 05000391/2008-003; 04/01/2008 - 06/30/2008; Watts Bar, Units 1 & 2; Problem Identification and Resolution and Event Followup.

The report covered a three-month period of routine inspection by the resident inspectors and a senior reactor inspector and two reactor inspectors. One NRC-identified Green finding and one self-revealing Green finding, both of which are non-cited violations (NCVs), were identified. The significance of an issue is indicated by its color (Green, White, Yellow, Red) using the Significance Determination Process in Inspection Manual Chapter 0609, Significance Determination Process (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green A Green, NRC-identified non-cited violation of Technical Specification 3.0.4.a was identified for entering Modes 2 and 1 without an operable channel of auxiliary feedwater automatic start on a trip of all main feedwater pumps as required by TS 3.3.2. The licensee defeated this channel by introducing a signal that artificially indicated that a main feedwater pump was operating. This practice existed since initial plant startup. The licensee entered this issue into their corrective action program as Problem Evaluation Report 147351.

  The finding is more than minor because it is associated with the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability of systems that respond to initiating events. Using IMC 0609, Appendix 0609.04, the finding was determined to be of very low safety significance because the finding did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time since other initiation signals were available to automatically start the auxiliary feedwater pumps if needed. The cause of the finding was directly related to the Implementation of Corrective Actions aspect in the Problem Identification and Resolution cross-cutting area, in that, the licensee failed to take appropriate corrective action in a timely manner to address the non-cited violation issued in NRC Inspection Report 05000390/2006004 associated with making plant mode changes with the auxiliary feedwater automatic start function trip of all main feedwater pumps inoperable (P.1(d)). (Section 4OA2.2)

  Green. A Green, self-revealing non-cited violation of Technical Specification 3.3.2 was identified for failure to have two trains of safety injection (SI) automatic actuation logic and two trains of feedwater isolation actuation logic operable while in Mode 3. Upon the removal of temporary jumpers, the relay which blocks the actuation circuitry from performing their function was not reset. This condition existed until approximately 12 hours later when the licensee reset the relay by closing the reactor trip breakers. The licensee entered this event into their corrective action program as Problem Evaluation Report 140641.

This finding is more than minor because it affected the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events and adversely affected the cornerstone's equipment performance attribute for availability and reliability. A Phase 2 evaluation in accordance with IMC 609, Significance Determination Process, determined the finding to be of very low safety significance (Green) because of the low decay heat at the end of a refueling outage; the time for operators to take recovery actions; and due to the plant conditions, only the containment high pressure SI actuation portion of the automatic SI actuation logic was affected. The cause of the finding was directly related to the documentation, procedures and component labeling cross-cutting aspect in the resources component of the Human Performance cross-cutting area, in that, the instructions used by personnel to remove the temporary jumpers failed to provide necessary steps to ensure the actuation logics were returned to an operable status (H.2(c)). (Section 4OA3)

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 action are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near 100 percent rated thermal power (RTP) until May 17, 2008, when the unit was removed from service to repair a hydrogen leak in the generator stator cooling system. The unit was returned to service on May 25, 2008. On June 6, 2008, the unit was ramped down to 45 percent RTP to replace the reactor coolant system (RCS) Loop II, Channel I Flow Transmitter. While returning the unit to 100 percent RTP on June 8, 2008, the number 3 Heater Drain Tank Level Controller malfunctioned causing the unit to runback to 75 percent RTP. The controller was replaced and the unit returned to 100 percent RTP on June 10, 2008. The unit remained at or near 100 percent RTP until the end of the inspection period.

Restart of construction on Unit 2 began in December of 2007. Information on Watts Bar Unit 2 reactivation can be found at http://www.nrc.gov/reactors/plant-specific-items/watts-bar.html

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather

a. Inspection Scope

Inspectors verified plant features, interviewed control room personnel, and reviewed procedures for operation and continued availability of offsite and alternate AC power systems and determined they were appropriate. Inspectors reviewed the licensee's procedures and interface agreements affecting these areas and the communications protocols between the northeast area dispatcher and the control room to verify that the appropriate information is exchanged when issues arise that could impact the offsite power system and the alternate AC power system. Documents reviewed are listed in Section 1R01 of the Attachment to this report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdowns

a. Inspection Scope

The inspectors conducted three equipment alignment partial walkdowns, listed below, to evaluate the operability of selected redundant trains or backup systems 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 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.

- Walkdown of 1A, 2A, and 2B emergency diesel generators (EDGs) while 1B out of service (OOS) for routine maintenance
- Walkdown of vital battery V when aligned to 125 vdc vital battery board II
- Walkdown of all EDGs while D common station service transformer OOS for water intrusion

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection – Walkdowns

a. Inspection Scope

The inspectors conducted tours of nine 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, 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.

- A-train residual heat removal (RHR) pump room
- B-train RHR pump room
- A-train containment spray (CS) pump room
- B-train CS pump room
- Turbine-driven auxiliary feedwater room
- A-train centrifugal charging pump (CCP) room
- B-train CCP room
- A-train safety injection pump (SIP) room
- B-train SIP room

b. Findings

No findings of significance were identified.

.2 Fire Protection - Drill Observation

b. Inspection Scope

On May 30, 2008, the inspectors observed an announced fire drill performed at the Unit 2 main bank transformers. The drill was observed to evaluate the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were: (1) specified number of individuals responded; (2) proper wearing of turnout gear; (3) self-contained breathing apparatus available and properly worn and used; (4) control room personnel followed procedures for verification and initiation of response; (5) fire brigade leader exhibited command and had a copy of the pre-fire plan; (6) fire brigade leader maintained control starting at the dress-out area; (7) fire brigade response timely and followed the appropriate access route; (8) control/command set up near the location and communications were established; (9) proper use and layout of fire hoses; (10) fire area entered in a controlled manner; (11) sufficient fire fighting equipment brought to the scene; (12) search for victims and propagation of the fire into other plant areas; (13) utilization of pre-planned strategies; (14) adherence to the pre-planned drill scenario and drill objectives acceptance criteria were met; and (15) fire fighting equipment returned to a condition of readiness to respond to an actual fire.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On June 3, 2008, the inspectors observed the simulator evaluations for Operations Shift Crew 2 per 3-OT-SRT-ECA3-1A Revision 1, Faulted-Ruptured Steam Generator. The plant conditions led to a Site Area Emergency level classification. The inspectors specifically evaluated the following attributes related to the operating crews' performance:

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of Abnormal Operating Instructions (AOIs) and Emergency Operating Instructions (EOIs)
- Timely and appropriate Emergency Action Level declarations per Emergency Plan Implementing Procedures (EPIP)
- Control board operation and manipulation, including high-risk operator actions
- Command and Control provided by the unit supervisor and shift manager

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

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two 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).

- Essential Raw Cooling Water (ERCW) returned to Category a(2) based on effectiveness monitoring of flushing program
- Problem Evaluation Report (PER) 136482, Shutdown boardroom chillers a(1) performance improvement plan

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors evaluated, as appropriate for the five 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 unforeseen 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.

- Maintenance risk associated with corrective maintenance on the 1A hydrogen analyzer during a B-train work week
- Maintenance risk associated with the A-train electric boardroom (EBR) chiller component outage
- Maintenance risk associated with the 2B EDG maintenance outage
- Maintenance risk associated with emergent voltage control problems with CSST D x-winding (B-train) while A-train EBR chiller OOS for maintenance
- Emergent inoperability of 1A motor-driven auxiliary feedwater caused reschedule of 1B EDG surveillance and maintenance on B-shutdown boardroom chiller

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 the compensatory measures, if involved, were in place, would work as intended, and were appropriately controlled; (4) where continued operability was considered unjustified, the impact on TS limiting condition for operations (LCOs) and the risk significance in accordance with the SDP. The inspectors verified that the operability evaluations were performed in accordance with SPP-3.1, Corrective Action Program.

- PER 139715, Reactor cavity blowout panel partially obstructed
- PER 142419, TDAFW pump room DC fan temperature switch
- PER 134494, A and B-train shutdown boardroom (SDBR) chiller's ERCW temperature control valves
- PER 145601, SDBR chill water pump B discharge check valve failed to close
- PER 144393, Non Class 1E annunciator circuits connected to Class 1E AFW Aux Control Room Transfer Switches

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary plant modification against the requirements of SPP-9.5, Temporary Alterations, and SPP-9.4, 10 CFR 50.59 Evaluation of Changes, Test, and Experiments, and verified that the modifications did not affect system operability or availability as described by the TS and UFSAR. In addition, the inspectors verified that: (1) the installation of the temporary modification was in accordance with the work package; (2) adequate configuration control was in place; (3) procedures and drawings were updated; and (4) post-installation tests verified operability of the affected systems.

- WO 08-815579-000, Leads lifted to separate Class 1E from non-Class 1E circuits for the Aux Control Room AFW transfer switch input to their respective annunciators.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed seven post-maintenance test procedures and/or test activities, (listed below) as appropriate, for selected risk-significant 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 05-823741, Annual polar crane maintenance MI-271.004
- WO 08-814711, Vital Battery Charger II high output voltage
- WO 08-814709, Vital Battery II, one cell with low individual cell voltage
- WO 07-816967, VT exams following ERCW flush of backup cooling line to 1A CCP
- WO 07-815967-000, Functional check of O-TS-065-0016-A, A-train EGTS heater temperature cutout switch
- WO 08-815683, Replacement of 1-PS-3-156-A
- WO 08-816327, Leak test on 1B CCP oil cooler

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The licensee began a forced outage on May 17, 2008, after manually tripping Unit 1 due to a hydrogen leak into the main generator stator cooling system. The inspectors observed portions of the shutdown, maintenance activities, and startup activities to verify that the licensee maintained defense-in-depth (DID) commensurate with the applicable TS. The inspectors monitored licensee controls over the outage activities listed below.

- Licensee configuration management, including daily outage reports, to evaluate DID and compliance with the applicable TS when taking equipment OOS.
- Installation and configuration of reactor coolant instruments to provide accurate indication and an accounting for instrument error.
- Controls over the status and configuration of redundant safety systems to ensure risk was minimized.
- Decay heat removal processes to verify proper operation and that steam generators, when relied upon, were a viable means of backup cooling.
- Heatup and startup activities to verify that TS, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes were met prior to changing modes or plant conditions.
- Reactor coolant system (RCS) integrity was verified by reviewing RCS leakage calculations, and containment integrity was verified by reviewing the status of containment penetrations and containment isolation valves.

b. Findings

A Green, NRC-identified NCV is discussed in Section 4OA2.3. No other findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed five 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.

Routine Surveillance Tests:

- WO 07-822414: 1-SI-30-26-A, Containment air return fan 1A-A quarterly operability test
- WO 08-811579: 1-SI-215-52-B, Diesel generator 1B-B 60 month performance test and battery charger test

In-Service Tests:

- WO 07-823635: 1-SI-63-901-B, Safety injection pump 1B-B quarterly performance test (QPT)
- WO 08-810173-000: 1-SI-62-901-A, CCP 1A-A QPT
- WO 08-810255-000: 1SI-74-901-A, RHR pump 1A-A QPT

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed one licensee-evaluated emergency preparedness drill and one simulator exercise to verify that the emergency response organization was properly classifying the event in accordance with Emergency Plan Implementing Procedure (EPIP)-1, Emergency Plan Classification Flowchart, and making accurate and timely notifications and protective action recommendations in accordance with EPIP-2, Notification of Unusual Event; EPIP-3, Alert; EIPIP-4, Site Area Emergency; EPIP-5, General Emergency; and the Radiological Emergency Plan. In addition, the inspectors verified that licensee evaluators were identifying deficiencies and properly dispositioning performance against the performance indicator criteria in Nuclear Energy Institute 99-02, Regulatory Assessment Performance Indicator Guideline.

- Quarterly training drill: Loss of offsite power followed by a loss of coolant accident leads to a General Emergency
- Simulator exercise: Faulted and ruptured steam generator

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verifications

The inspectors sampled licensee submittals for the two PIs listed below. To verify the accuracy of the PI data reported for the period April 1, 2007, through March 31, 2008, the inspectors reviewed licensee records and interviewed cognizant personnel. The PI definitions and guidance contained in NEI 99-02, Regulatory Assessment Indicator Guideline, Revision 5, were used to verify the basis in reporting for each data element.

Mitigating Systems Cornerstone PI

- Safety system functional failures

Barrier Integrity Cornerstone PI

- RCS leak rate

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered into the Corrective Action Program

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

.2 Annual Sample: Corrective actions associated with NCV 05000390/2007004-01, Failure to Promptly Correct an Identified Procedural Deficiency Prior to Subsequent Maintenance

a. Inspection Scope

The inspectors reviewed the plan and implementation of corrective actions for NCV 05000390/2007004-01, which were documented in PERs 111386 and 125404.

b. Findings and Observations

No findings of significance were identified. However, the inspectors identified several observations which were discussed with the licensee. PERs 111368 and 125404 had one common corrective action to revise licensee procedure MI-57.029, HFA Relay Maintenance, to add an electrical bench test that would measure the closed contact resistance of the relay in both the de-energized and energized/latched state to ensure the closed contact resistance is less than one ohm.

The inspectors reviewed a completed work package that included the new revision of MI-57.029 that was part of WO 2003-016476-016. The WO set up and tested two separate HFA relays. The inspectors determined the MI-57.029 procedure revision was not performed as written for one of the relays. The relay in question had six sets of contacts; four normally-closed contacts and two normally-open contacts. With the relay in the energized state, the resistance readings for two normally-open contacts were less than one ohm, as expected. With the relay in the de-energized state, resistance readings of less than one ohm were recorded for all six sets of contacts; this was not expected and was not identified during the work order closeout review. The observations were considered minor because the resistance checks were considered an enhancement to the procedure and the associated relay's post maintenance testing was completed satisfactorily.

The inspectors discussed the observations with the licensee. The licensee initiated PER 147461 to address the observations.

.3 Annual Sample: Corrective actions associated with NCV 05000390/2006004-04, Failure to Have an AFW Autostart Signal on Loss of All MFW Pumps During Plant Startup

a. Inspection Scope

The inspectors reviewed the corrective actions taken in response to NCV 05000390/2006004-04. This review included reviewing whether the problem was correctly identified and evaluated and whether corrective actions taken were effective in addressing the issue. Specifically, the inspectors reviewed implementation of the licensee's corrective actions during the March 2008 restart from a refueling outage. The inspectors also reviewed PER 110770 associated with this NCV.

b. Findings and Observations

Introduction: A Green, NRC-identified non-cited violation of TS 3.0.4.a was identified for entering Modes 2 and 1 without an operable channel of auxiliary feedwater automatic start on a trip of all main feedwater pumps as required by TS 3.3.2. The licensee defeated this channel by introducing a signal that artificially indicated that a main feedwater pump was operating. This practice existed since initial plant startup. The licensee entered this issue into their corrective action program as Problem Evaluation Report 147351.

Description: TS 3.3.2 item 6e required that the AFW automatic start function on the trip of all main feedwater pumps be operable in Modes 1 and 2. This function allowed AFW flow to the steam generators in case the main feedwater pumps trip during power operation. Neither the plant design nor TSs provides allowance for defeating this start signal during startup and low power operations. However, the licensee had recognized since original startup that operating the turbine driven AFW pump might make reactivity control difficult during startup and low power operations. As a result, the licensee had been defeating this AFW automatic start function by resetting the trip circuit on a main feedwater pump such that the AFW automatic start circuit would have indication that a main feedwater pump was running, even though it was not.

In 2004, the licensee had submitted a TS amendment to allow bypassing this function. However, due to NRC questions on the technical bases for the specific change requested by the licensee, the licensee decided to withdraw the request. In 2006, the NRC issued NCV 05000390/2006004-04 for failing to have an operable AFW start signal on loss of all main feedwater pumps during startup operations. As corrective actions for the NCV, the licensee changed operating procedures to use TS 3.0.4b which allows, based on the results of a risk assessment, entering a Mode when a TS LCO is not met. Prior to the refueling outage, the inspectors informed the licensee that use of TS 3.0.4b in their procedure was not consistent with their own procedural restrictions on the use of TS 3.0.4b. As a result, during startup from a refueling outage in March 2008, the licensee relied upon their position of what constituted an operable channel of AFW automatic actuation circuitry. Their position was that if a channel would process an input signal correctly, then the channel was operable, even when the input signal was known not to be valid, i.e., the input did not represent the correct status of the equipment. In this case, the licensee reset a main feedwater pump trip condition such that the AFW automatic start circuit would have indication that the main feedwater pump was running, even when it was not running. This operating practice was not described in the UFSAR. After the startup, based upon a review of this position by the Technical Specification Branch in the Office of Nuclear Reactor Regulation, the licensee was informed that their position on an operable channel was inconsistent with their licensing basis and that TS requirements associated with entering Modes 1 and 2 had not been met.

Analysis: Failure to operate the plant in accordance with TS was a performance deficiency. The finding is more than minor because it is associated with the configuration control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability of systems that respond to initiating events. The cornerstone was impacted since the licensee operated outside their licensing basis in Modes 1 and 2 without an AFW automatic start signal for a loss of MFW. The finding was determined to be of very low safety significance because the finding did not represent a loss of system safety function, because the AFW automatic start on low-low steam generator level, loss of offsite power, and safety injection were functional.

The cause of the finding was directly related to the Implementation of Corrective Actions aspect in the Corrective Action Program component of the Problem Identification and Resolution cross-cutting area, in that, the licensee failed to take appropriate corrective action in a timely manner to address the NCV issued in NRC Inspection Report 05000390/2006004 associated with making plant mode changes with the AFW automatic start function trip of all MFW pumps inoperable (P.1(d)).

Enforcement: TS 3.0.4.a requires that entry into a Mode or other specified condition in the applicability shall only be made when the associated actions to be entered permit continued operation in the Mode to be entered for an unlimited period of time. Contrary to this, from initial power operations until March 2008, the licensee had entered Modes 2 and 1 with one MFW pump trip channel for the AFW automatic start function inoperable when the associated actions of TS 3.3.2 did not permit operation for an unlimited period of time. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as PER 147351, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 5000390/2008003-01, Plant Startup with Inoperable AFW Automatic Start on Trip of All MFW Pumps.

4OA3 Event Followup

(Closed) LER 05000390/2008-001-00, Automatic Safety Injection (SI) Actuation Instrumentation Blocked in Modes 4 and 3

The event is discussed in the Description paragraph below and Section 4OA7. The LER identified that the two key causes were an inadequate General Operating Instruction, and an inadequate Instrument Maintenance Instruction (IMI). Other cause evaluations also identified knowledge deficiencies and deficient use of annunciator response procedures as contributing causes. The inspectors verified that the corrective actions and extent of condition were consistent with the root cause(s). As documented below, one self-revealing and one Licensee-Identified NCV were identified. This LER is considered closed.

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the event described in the LER.

b. Findings

Introduction: A Green, self-revealing violation of TS 3.3.2 was identified for having both trains of SI Automatic Actuation Logic and Actuation Relays and Turbine Trip and Feedwater Isolation Actuation Logic and Actuation Relays inoperable in Mode 3. An unexpected annunciator for the current plant conditions alerted the operators that the automatic actuation logic circuitry had not be restored to operable status as they had thought when temporary jumpers had been removed earlier.

Description: On February 12, 2008, as part of the Cycle 8 refueling outage, temporary jumpers were installed to block the following functions of TS 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation:

- The automatic actuation of the safety injection system
- The automatic actuation of the feedwater isolation signal.

The jumpers were installed in accordance with IMI 99.040, Auto Safety Injection (SI) Block, Feedwater Isolation Block, and Maintain Source Range In Service Jumpers.

On March 21, 2008, while in Mode 3, plant personnel identified that these temporary jumpers had not been removed. The temporary jumpers were removed at 0958 EDT on March 21. Because the licensee identified the failure to remove the temporary jumpers the failure is dispositioned in Section 4OA7 as a licensee-identified NCV.

As reactor pressure was raised on March 21, the Low Steam Pressure Safety Injection Blocked Steam Pressure Rate Steam Line Isolation-Active (P-11) and Pressurizer Pressure Safety Injection Blocked annunciators cleared. However, the Auto SI Blocked annunciator remained on. The licensee subsequently determined that when the reactor trip breakers had been opened prior to the removal of the temporary jumpers, a relay that blocks the actuation circuitry from functioning had sealed-in. Reclosing the reactor trip breakers at 2206 EDT reset this blocking relay and restored the actuation functions to an operable status.

Analysis: Not closing the reactor trip breakers to reset the actuation logic after removing the temporary jumpers was a performance deficiency. This finding is more than minor because it affected the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events and it adversely affected the cornerstone's equipment performance attribute for availability and reliability.

The inspectors evaluated this finding using MC 0609, Appendix A, and determined that a significance determination process (SDP) Phase 2 evaluation was required since the finding represented an actual loss of safety function of both trains for greater than their TS-allowed outage time. Using the entire time from when the reactor was placed into Mode 4 with the temporary jumpers installed until the circuits were returned to operable status, a Phase 2 evaluation determined that the finding was of very low safety significance (Green). Assumptions used in the SDP included: (1) credit for operator recovery actions with a value of three due to very low decay heat following a refueling outage which extends the amount of time available for the operators to take recovery actions except for sequences with include both EHIP and EILP (high and low head Injection); and, (2) only those sequences associated with the unintended blocking of Containment High Pressure were evaluated because Low Steam Line Pressure and Low Pressurizer Pressure SI actuation circuits were blocked by the P11 permissive per plant design. The remaining sequences were associated with small, medium and large break loss of coolant accidents and stuck open pressurizer relief valves.

The cause of the finding was directly related to the documentation, procedures and component labeling cross-cutting aspect in the resources component of the Human Performance cross-cutting area, in that, the instructions used by personnel to remove the temporary jumpers failed to provide necessary steps to ensure the actuation logics were returned to an operable status (H.2(c)).

Enforcement: Unit 1 TS 3.3.2 requires, in part, that two trains of safety injection automatic actuation logic and actuation relays be operable in Modes 1, 2, 3 and 4. Additionally, it also requires two trains of feedwater isolation automatic actuation logic and actuation relays to be operable in Modes 1, 2, and 3. Contrary to this, on March 21, 2008, two trains of safety injection automatic actuation logic and actuation relays and two trains of feedwater isolation automatic actuation logic and actuation relays were not operable in Mode 3, in that, both trains for both functions were blocked from automatically actuating for approximately 12 hours. Because this failure to comply with Technical Specifications is of very low safety significance and has been entered into the CAP as PER 140641, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000390/2008003-02, Failure to Comply with Technical Specification 3.3.2 to Have Two Trains of Automatic Actuation Logic and Actuation Relays for Safety Injection and Feedwater Isolation Operable.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspector reviewed the status of the implementation of the licensee's actions in response to GL 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors, for Unit 1. The onsite inspection which verified the installation of GL 2004-02 related modifications was performed in November 2006 (NRC Report No.: 50-390/2006005). The licensee's GL 2004-02 commitments which were incomplete at the time of the inspection included modification of the cold leg injection orifice and resolution of discrepancies in the debris generation evaluation (PER 106299). Additionally, the licensee requested and received approval (Letter, USNRC to TVA, dated 12/6/07) for a completion extension date to the February 2008 outage to modify the min-K insulation in containment. The inspector requested information from the licensee to review the status of the incomplete commitment items and performed an in-office review during the week of May 5, 2008, to verify completion of the outstanding commitment items.

The inspector reviewed the licensee design and work documentation to verify that the above outstanding GL 2004-02 commitments were completed. Documents reviewed are listed in Section 4OA5 in the Attachment to this report.

b. Findings and Observations

No findings of significance were identified.

The licensee's corrective actions identified in their initial and supplemental responses to GL 2004-02 were complete. The implementation of plant modifications and procedure changes was consistent with the requirements of 10 CFR 50.59.

This documentation of TI-2515/166 completion as well as any results of sampling audits of licensee actions will be reviewed by the NRC staff (Office of Nuclear Reactor Regulation - NRR) as input along with the Generic Letter (GL) 2004-02 "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" responses to support closure of GL 2004-02 and Generic Safety Issue (GSI)-191 "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance." The NRC will notify each licensee by letter of the results of the overall assessment as to whether GSI-191 and GL 2004-02 have been satisfactorily addressed at that licensee's plant(s). Completion of TI-2515/166 does not necessarily indicate that a licensee has finished all testing and analyses needed to demonstrate the adequacy of their modifications and procedure changes. Licensees may also have obtained approval of plant-specific extensions that allow for later implementation of plant modifications. Licensees will confirm completion of all corrective actions to the NRC. The NRC will track all such yet-to-be-performed items identified in the TI-2515/166 inspection reports to completion and may choose to inspect implementation of some or all of them.

.3 (Closed) NRC TI 2515/172, Reactor Coolant System Dissimilar Metal Butt Welds (DMBWs) - Unit 1

a. Inspection Scope

From March 31 to April 4, 2008, the inspectors reviewed the licensee's activities related to the inspection and mitigation of DMBWs in the Reactor Coolant System (RCS) to ensure that the licensee activities were consistent with the industry requirements established in the Materials Reliability Program (MRP) document MRP-139, Primary System Piping Butt Weld Inspection and Evaluation Guidelines, July 2005. The inspections covered the following: a) implementation of baseline volumetric examinations for the six DMBWs on the pressurizer nozzles during the last outage; b) documentation review and direct observation of the volumetric examination and Mechanical Stress Improvement Process (MSIP) for the DMBWs on the pressurizer nozzles during this refueling outage; and c) review of the MRP-139 program. For the observations of Ultrasonic Examination (UT) prior to the application and the application of MSIP for the inservice inspection during the refueling outage, please refer to Section 1R08, Inservice Inspection (ISI) Activities of NRC Integrated Inspection Report 05000390/2008002. Documents reviewed are listed in Section 4OA5 in the Attachment to this report.

b. Findings and Observations

No findings of significance were identified. Specific observations to the TI inspection requirements are provided below.

A. <u>MRP-139 Baseline Inspections</u>

1) Have the baseline inspections been performed or are they scheduled to be performed in accordance with MRP-139 guidance? Were the baseline inspections of the pressurizer temperature dissimilar metal butt welds (DMBWs) of the nine plants listed in TI 2515/172, 03.01.b completed during the spring 2008 outages?

   Yes. The licensee performed baseline volumetric inspection activities with the conventional or manual Ultrasonic Examination (UT) in September 2006 for the six DMBWs on the pressurizer nozzles, required to be completed per MRP-139 Section 1.2. The licensee performed the profile measurement of the weld and pipe surfaces for the preparation of the baseline volumetric examination of UT and then performed the UT. For example, the licensee performed the baseline volumetric inspection documented in the UT Report R-1161 for Weld No. WP-10-SE, Pressurizer 15" Diameter Surge Line Dissimilar Metal, Nozzle-to-Safe End Weld, September 15, 2006. Therefore, the licensee met the implementation deadline requirement set on December 31, 2007, by the MRP-139 for the baseline volumetric examination by performing the UT on the six DMBWs on the pressurizer nozzles for Unit 1. The licensee used the conventional UT Procedure N-UT-82, "Generic Procedure for the Ultrasonic Examination of Dissimilar Pipe Welds," Rev. 2 for the baseline volumetric examination. The examination was performed using ASME Section XI, Appendix VIII, qualified techniques, equipment, and personnel. The procedure was qualified in accordance with ASME Section XI, Appendix VIII, as implemented through the EPRI Performance Demonstration Initiative (PDI) Program. The examinations resulted in 100% coverage in the circumferential and axial beam directions. The inspectors reviewed the procedures, work orders, work packages, examination reports, equipment qualification records, and personnel qualification and certificates.

   There are no Alloy 82/182 DMBWs greater than or equal to 4" nominal pipe size (NPS) and less than 14" NPS exposed to temperatures equivalent to the hot leg for Unit 1.

   There are no Alloy 82/182 DMBWs greater than 14" NPS exposed to temperatures equivalent to the hot and cold legs connecting to the Steam Generators for Unit 1 because of the use of stainless steel welds and piping for the recently replaced new Steam Generators.

   Alloy 82/182 DMBWs greater than 14" NPS exposed to temperatures equivalent to the hot and cold legs connecting to the Reactor Vessel for Unit 1 will be examined for the baseline volumetric examination and mitigated by using weld overlays or MSIPs in Fall 2009 and will meet the MRP-139 implementation deadline of December 31, 2009, for the hot leg temperature and December 31, 2010, for the cold leg temperature.

2) Is the licensee planning to take any deviations from MRP-139 requirements?

   No. The licensee has not taken any deviations or submitted any requests for deviations from MRP-139 requirements.

B. Examinations/ Weld Overlays Mitigations Performed (Not Applicable)

The licensee has not implemented weld overlays as a mitigation method for the DMBWs.

C. Examinations and Mechanical Stress Improvement

1) Are the nozzle, weld, safe end, and pipe configurations, as applicable, consistent with the configuration addressed in the stress improvement (SI) qualification report?

Yes. They were consistent with the qualification reports. Westinghouse and its vendor, NuVision Engineering, performed three analytical verifications of MSIP to be used for all the dissimilar welds on the pressurizer nozzles. For example, the qualification report for the surge nozzle was Westinghouse Document 4387-4-001-01, Analytical Verification of MSIP for Pressurizer Surge Nozzle to Safe End Weld, Watts Bar Units 1 and 2, December 2007, Rev. 1. The inspectors reviewed this document, compared it to the procedure used, and observed the implementation of the MSIP during this outage between February 19 - 22, 2008.

2) Does the SI qualification report address the location radial loading is applied, the applied load, and the effect that plastic deformation of the pipe configuration may have on the ability to conduct volumetric examinations?

Yes. The analytical verification provided the loading location and the applied load and also considered the effect that plastic deformation of the pipe configuration may have on the ability to conduct volumetric examinations. The licensee completed the UT after each MSIP and obtained 100 percent coverage in both axial and circumferential directions for the six DMBWs on the pressurizer nozzles for Unit 1.

3) Do the licensee's inspection procedure records document that a volumetric examination per the ASME Code, Section XI, Appendix VIII was performed prior to and after the application of the SI?

Yes. The licensee performed a volumetric examination of UT and recorded the result prior and after the application of the MSIP for all six DMBWs on the pressurizer nozzles per the ASME Code, Section XI, and Appendix VIII. For example, the licensee performed the UTs prior to and after the MSIP for the surge nozzle documented in the UT Report R-1157 for Weld No. WP-10-SE, Pressurizer 15" Diameter Surge Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 20, 2008 for the examination prior to the MSIP and UT Report R-1558 for Weld No. WP-10-SE, Pressurizer 15" Diameter Surge Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 22, 2008 for the examination after the MSIP. The licensee used the Procedure N-UT-82, which was used for the baseline volumetric examination, for the volumetric examinations prior to and after the MSIP. The examination was performed using ASME Section XI, Appendix VIII, qualified techniques, equipment, and personnel. The procedure was qualified in accordance with ASME Section XI, Appendix VIII, as implemented through the EPRI Performance Demonstration Initiative (PDI) Program. The examinations resulted in 100% coverage in the circumferential and axial beam directions. The inspectors reviewed the procedures, work orders, work packages, examination reports, equipment qualification records, and personnel qualification and certificates.

4) Does the SI qualification report address limiting flaw sizes that may be found during pre-SI and post-SI inspections and that any flaws identified during the volumetric examination are to be within the limiting flaw sizes established by the SI qualification report.

   Yes. Westinghouse Project Plan No.PP-A600-07-6, Watts Bar Pressurizer Nozzle Mechanical Stress Improvement Process (MSIP), Rev. 0, Dated March 17, 2005, stated that the flaw is limited to the smaller than 30 percent through wall, or 10 percent for a circumferential crack Pre and Post-MSIP. Westinghouse Report, WCAP-9443-R1-S1, Supplement 1, Watts Bar Units 1 and 2 Evaluation of the Pressurizer Surge Line Piping from the Application of the Mechanical Stress Improvement Process (MSIP) on the Pressurizer Surge Nozzle, Rev. 2, Dated January, 2008 provided more details for the flaw analysis and limitation on the surge nozzle. There were no cracks or flaws identified by the UTs for Pre or Post-MSIPs for all six DMBWs on the pressurizer nozzles.

5) Performed by qualified and knowledgeable personnel?

   Yes. The personnel such as operators for the MSIP and conventional UT examiners involved in the DMBWs on the pressurizer nozzles for Unit 1 for the mitigation using MSIP were qualified and knowledgeable in accordance with the requirements of MRP-139 and the ASME Code. The examiners were qualified Level II or Level III in the UT methods as required by the UT procedures and in accordance with the vendor's written practice for NDE personnel. The UT examiners were also PDI-qualified for the specific UT procedure they implemented. The final examination reports were reviewed by the vendor and/or licensee Level III UT examiners. The inspectors reviewed the operator performance qualification records and compared them with the requirements of the vendor procedures and the ASME Code for the MSIP application.

6) Performed such that deficiencies were identified, dispositioned, and resolved?

   Yes. The licensee and vendors performed the UTs and MSIPs based on the procedures which can identify, disposition, and resolve the problems.

   During the application of the MSIP for the six DMBWs on the pressurizer nozzles, the MSIP clamp for the spray nozzle was misplaced approximately 0.5" axially closer to the pressurizer. The licensee issued Problem Evaluation Report (PER) 139019, Pressurizer Spray Line Weld MSIP Position Correction to address the problem. The vendor, Westinghouse, issued Corrective Action Process (CAP) Temporary Issue Report WAT-MSIP-2008-001 and Issue Report 08-056-W006, Watts Bar Pressurizer Spray Nozzle Mechanical Stress Improvement Process (MSIP) Performed in the Wrong Location. The vendor, NuVision Engineering, performed a Supplemental Analysis to Analytical Verification of MSIP for Spray Nozzle to Safe End for Watts

Bar Unit 1 on February 27, 2008 to evaluate the problem for the misplaced location and accepted the result. Westinghouse also performed an Addendum to the Watts Bar Units 1 and 2 Pressurizer Stress Report WNET-130 Vol. 1, Rev. 3 in February 2008 to evaluate the effect of misplacement of the MSIP Clamp on Fatigue of the Pressurizer Spray Nozzle for Watts Bar Unit 1. This Addendum was used to verify that the existing ASME qualification of the MSIP remains valid.

D. In-service Inspection Program

1) Has licensee prepared an MRP-139 in-service inspection program?

No. The licensee did not have a stand-alone MRP-139 in-service inspection program document. However, the licensee's MRP-139 inservice inspection program is included in the ASME Section XI In-service Inspection Program (ISI Program). The inspectors reviewed the Watts Bar Second Interval ISI Plans for Unit 1. The licensee will revise the Second Interval ISI Plans to add more details to the examination methods and frequencies for the MRP-139 ISI requirements.

2) Are welds appropriately categorized?

Yes. The welds were appropriately categorized by the licensee responsible engineer. However, the licensee is in the process to integrate the requirements of the examination methods and frequencies for all DMBWs in the Second Interval ISI Plans for Unit 1 to meet the MRP-139 ISI requirements. The licensee has enough time to perform this task. The inspectors reviewed all DMBWs categorized at the time of the inspection for appropriate categorization.

3) Are inspection frequencies consistent with the requirements of MRP-139?

Yes. The licensee plans inspection frequencies for welds in the MRP-139 ISI program to be consistent with the requirements of MRP-139.

4) What is the licensee's basis for categorizing welds as H or I and plans for addressing potential PWSCC?

No welds were categorized as Categories H or I.

5) What deviations has the licensee incorporated and what approval process was used?

No deviations to MRP-139 have been incorporated by the licensee.

.4 (Closed) Temporary Instruction 2515/174, Hydrogen Igniter Backup Power Verification

a. Inspection Scope

The objective of TI 2515/174, "Hydrogen Igniter Backup Power Verification," was to verify that licensees have adequately implemented commitments related to providing backup power to containment hydrogen igniters.

During this inspection period, the inspectors reviewed the licensee's response to GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion during a Severe Accident." Their response included a commitment to revise the back-up generator procedure(s) to include supplying one train of containment hydrogen igniters per unit, and train personnel to the procedure revision, by December 31, 2007.

The inspector's review of the licensee's commitments consisted of the review and verification that appropriate procedures had been established to govern the provision of backup power to the igniters and included appropriate cautions. Additionally, the feasibility and availability of the power supply to be transported and/or connected from its storage location to the access location for providing power to the igniters using readily available equipment (i.e. fitting, cables, tool, etc.) was evaluated. The inspector also reviewed electrical schematics and capacity/loading calculations to verify that the rating of the portable power supply was adequate to continuously power at least one train of igniters and that unnecessary portions of the internal power distribution system could be separated from portions necessary to provide power to the igniters from the portable supply. The licensee's training records and related vendor documents were all reviewed to verify that a suitable training program had been established to train staff in the actions necessary to provide back up power and to verify that all necessary equipment is being tested, stored and maintained consistent with vendor recommendations. Documents reviewed are listed in Section 4OA5 in the Attachment to this report.

b. Findings and Observations

Evaluation of Inspection Requirements

No findings of significance were identified. Specific observations to the TI inspection requirements are provided below.

(1) Did the licensee modify plant equipment and implement training programs and procedures to provide backup power to at least one complete train of hydrogen igniters?

Yes. Watts Bar procured a trailer-mounted diesel generator set with a capacity of 2 megawatts (MW). The generator can be connected to feed the 6.9kV Shutdown Boards through temporary cables by connecting to the 6.9kV Unit Boards in the turbine building. Mitigating Actions MA-1, Recovery from Loss of Shutdown Power and Loss of ERCW procedure provided this guidance and would normally be entered as a result of a loss of emergency raw cooling water (ERCW) in conjunction with a loss of shutdown power.

The portable 2MW generator was in a stationary position. If routing cables from the 2MW diesel generator to the 6.9kV Unit Boards were not feasible, a 65kW diesel generator located in the Auxiliary Building could be used to provide power directly to the igniters. The 2MW generator capability along with the proper connections would enable at least one train of hydrogen igniters to be restored once the need has been determined.

Operators and select maintenance staff were trained on applicable procedures required to facilitate provision of backup power.

(2) Did the licensee provide the equipment necessary to provide backup power to the hydrogen igniters?

The trailer-mounted diesel generator set provided backup power to the hydrogen igniters through the C and A Vent Boards. All necessary cables and fittings were pre-staged at the storage location of the diesel generator and other designated locations.

The inspector was unable to determine, by official record, that the movement of the power supply and connection of necessary fittings and cables to provide backup power to the igniters could be completed within three hours. Additional information was received that showed training and timing achievements gathered in 2004 when the 2MW diesel generator was first procured. This information was part of the personal notebook belonging to the project manager for that initial installation and testing and was not an official record. The licensee responded that because this issue was beyond the design basis, components and activities were not treated as safety-related or under the quality assurance program. Hence, no official documentation was required, and none was generated. Based on the additional information provided, the inspector concluded the timeline could be met.

(3) Did the licensee establish appropriate procedures to govern the provision of backup power to the igniters?

The licensee modified their mitigating actions procedure, MA-1, to include governance of the provision of backup power to the igniters; however, revisions to the procedure did not include cautions against actuation of the igniters after indications of severe core damage are present. The inspector reviewed an open PER 145726 which the licensee had initiated to document that current implementing procedures required improvement. The licensee was in the process of adding to the mitigating procedure a caution note regarding actuation of the igniters under severe core damage conditions.

(4) Did the licensee establish a suitable training program to train selected staff in the actions necessary to provide backup power to the igniters?

The inspector reviewed and verified training and development materials and attendance records. Training on the actions necessary to provide backup power to the igniters was included in the licensee's B.5.b training that selected staff received during the 4th quarter of 2007. Training was in place with regard to providing backup power and this training was commensurate with SAMG (Severe Accident Mitigation Guidelines) training, which was conducted on a four year frequency. To date, personnel trained included Shift managers, Unit Supervisors, Unit Operators, Assistant Unit Operators, Operations Training Personnel, Maintenance Shift Supervisors, and Fire Protection Ops personnel. It had not yet been determined if training will be extended to include Site Emergency Directors, the Operations Manager, the Operations Support Center Manager, and the Technical Analysis Manager. This was to be addressed in future B.5.b assessments (PER 145731).

Training conducted as of December 2007 addressed the use of emergency guidelines; however, no drills or dry runs through the Mitigating Actions procedures have been conducted since 2004. There is no requirement detailing that drills or dry runs for this procedure must be completed; however the licensee indicated that drills may be added in the future to demonstrate the ability to perform the strategies within the required timeframe. This was to be addressed in future B.5.b assessments (PER 145731).

(5) Did the licensee establish maintenance and testing schedules consistent with vendor recommendations for permanently installed equipment?

The contract vendor performed inspection and maintenance on the Blackout Diesel Generators as a part of the licensee's preventive maintenance work. Inspection and maintenance work was performed on a twelve week frequency. The licensee performed routine surveillance on the permanently installed hydrogen mitigation system consistent with Technical Specifications.

Although the 2MW Diesel Generator was tested by a regularly scheduled preventive maintenance, the companion transformer was not tested and no routine maintenance was performed. The licensee stated that the vendor manual maintenance requirements were intended for a transformer that has continuous use. The manual specified: "It is suggested that a maintenance check be carried out after the first three months of operation and then at twelve month intervals after that. This interval may be increased or decreased according to site conditions." The licensee took the position that since the transformer did not have three months of operation; the vendor manual requirements were not applicable for their application. However, PER 145746 was initiated to evaluate the need to improve control of preventive maintenance for such equipment.

4OA6 Meetings, including Exit

.1 Exit Meeting

The inspectors presented the inspection results to Mr. M. Skaggs and other members of licensee management on July 7, 2008. Another exit was conducted with

Mr. M. Brandon on August 5, 2008. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Annual Assessment Meeting Summary

On April 7, 2008, the NRC's Chief of Reactor Project's Branch 6 and the Senior Resident Inspector assigned to the Watts Bar Nuclear Plant met with TVA to discuss the NRC's Reactor Oversight Process and the Watts Bar Unit 1 annual assessment of safety performance for the period of January through December 2007. The major topics addressed were: the NRC's assessment program, the results of the Watts Bar assessment, and NRC security activities. Attendees included Watts Bar site management, members of site staff, and corporate management.

This meeting was open to the public. The presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML081300659. ADAMS is accessible from the NRC Web site at http://www/nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

4OA7 Licensee-Identified Violations

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

TS 3.3.2 requires that the ESFAS instrumentation for each function in Table 3.3.2-1 shall be operable. Table 3.3.2-1 function 1b, required two trains of SI automatic actuation logic and actuation relays be operable in Modes 3 and 4. Table 3.3.2-1 Function 5a, required two trains of turbine trip and feedwater isolation automatic actuation logic and actuation relays be operable in Mode 3. Contrary to this, on March 20 and 21, 2008, two trains of SI automatic actuation logic were not operable, in that, both trains were blocked. Also on March 21, two trains of SI and feedwater isolation automatic actuation logic were not operable, in that, both trains were blocked. This finding is more than minor because it affected the Mitigating Systems Cornerstone objective and adversely affected the cornerstone's equipment performance attribute for availability and reliability. A phase 2 SDP evaluation determined that the finding was of very low safety significance.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

L. Belvin, Radiation Protection Manager
G. Boerschig, Plant Manager
M. Brandon, Licensing and Industry Affairs Manager
D. Holmes, Senior Engineering Specialist for Alloy 600 and MRP-139
F. Leonard, Engineering
J. Lockwood, ISI Coordinator
K. Lovell, Maintenance and Modifications Manager
G. Mauldin, Site Engineering Manager
M. McFadden, Site Nuclear Assurance Manager
A. Scales, Operations Manager
M. Skaggs, Site Vice President
S. Smith, Operations Superintendent
D. Voeller, Site Support Manager
M. Welch, ISI Level III

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-390/2008003-01	NCV	Plant Startup with Inoperable AFW Automatic Start on Trip of All MFW Pumps (Section 4OA2.2)
50-390/2008003-02	NCV	Failure to Comply with Technical Specification 3.3.2 to Have Two Trains of Automatic Actuation Logic and Actuation Relays for Safety Injection and Feedwater Isolation Operable (Section 4OA3)

Closed

05000390/2008-001-00	LER	Automatic Safety Injection (SI) Actuation Instrumentation Blocked in Modes 4 and 3 (Section 4OA3)
2515/166	TI	Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) Unit 1 (Section 4OA5.2)
2515/172	TI	Reactor Coolant System Dissimilar Metal Butt Welds (DMBWs) - Unit 1 (Section 4OA5.3)
2515/174	TI	Hydrogen Igniter Backup Power Verification (Section 4OA5.4)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

IGA 6, Intergroup Service Agreement for Transmission/Power Supply
TRO-TO-SOP-10.130, Watts Bar Nuclear Plant (WBN) Grid Operating Guide
SPP-7.1, On Line Work Management
TI-12.15, 161KV Offsite Power Requirements

Section 4OA5 Other: TI 2515/166

Letter, USNRC toTVA, GL 2004-02, Watts Bar Nuclear Plant, Unit 1, - Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," Extension Request Evaluation, dated 12/6/07
DCN 52057, Replace Flow Element 1-FE-63-170 with More Restrictive Orifice, dated 1/10/07
PER 106299, Non-Conservative Inputs in debris generation calculation, initiated 7/5/07
DCN 52226, Installation of Restraint Bands on Installed Containment Min-K Insulation, dated 4/9/07
PIC 52314-A, Replacement Steam Generator Insulations – Staging DCN, dated 11/27/06

Section 4OA5 Other: TI 2515/172

Procedures and Specifications

EPRI Materials Reliability Program (MRP) 139, Primary System Piping Butt Weld Inspection and Evaluation Guideline, July 14, 2005
EPRI MRP-126, Generic Guidance for Alloy 600 Management
EPRI MRP-121, Mechanical Stress Improvement Process (MSIP) Implementation and Performance Experience for PWR Applications
Business Practice BP-257, Integrated Material Issues Management Plan, Rev. 7
1-TRI-0-10.2, ASME Section XI ISI/NDE Program, Rev. 000
N-UT-82, Generic Procedure for the Ultrasonic Examination of Dissimilar Pipe Welds, Rev. 2
Westinghouse Procedure IR-A600-07-1, Interface Requirements for Watts Bar Unit 1 Mechanical Stress Improvement Process (MSIP), Rev. 0
Westinghouse Procedure MRS-SSP-2184, Watts Bar Unit 1 Pressurizer Mechanical Stress Improvement Process (MSIP), Rev. 1
Westinghouse MSIP Parameters for Pressurizer Surge Nozzle to Safe End Weld
NuVision Engineering Procedure MSIP for Toling Loading Applied to Nozzle Safe Ends
NuVision Engineering Mechanical Stress Improvement Process (MSIP) Overview

Corrective Action Documents

Problem Evaluation Report (PER) 139019, Pressurizer Spray Line Weld MSIP Position Correction
PER 140891, CRP-ENG-SS-08-003, Augmented ISI Program Examination Documentation

Other Records

NRC Memorandum for Harold R. Denton, Director of Office of Nuclear reactor regulation from Eric S. Beckjord, Director of Office of Nuclear Regulatory Research, Research Information Letter No. 149 – Evaluation of the Mechanical Stress Improvement Process

Westinghouse Document 4387-4-001-01, Analytical Verification of MSIP for Pressurizer Surge Nozzle to Safe End Weld, Watts Bar Units 1 and 2, December 2007, Rev. 1

Westinghouse Document 4387-4-002-01, Analytical Verification of MSIP for Pressurizer Safety and Relief Nozzle to Safe End Weld, Watts Bar Units 1 and 2, December 2007, Rev. 1

Westinghouse Document 4387-4-003-01, Analytical Verification of MSIP for Pressurizer Spray Nozzle to Safe End Weld, Watts Bar Units 1 and 2, December 2007, Rev. 1

Westinghouse Report, WCAP-9443-R1-S1, Supplement 1, Watts Bar Units 1 and 2 Evaluation of the Pressurizer Surge Line Piping from the Application of the Mechanical Stress Improvement Process (MSIP) on the Pressurizer Surge Nozzle, Rev. 2, Dated January, 2008

Westinghouse Project Plan No. PP-A600-07-6, Watts Bar Pressurizer Nozzle Mechanical Stress Improvement Process (MSIP), Rev. 0, Dated March 17, 2005

NUVision Engineering Document 4387-214-001-00, MSIP Parameter Mechanical Stress Improvement Process (MSIP) for Pressurizer Surge Nozzle to Safe End Weld, January 2008

NuVision Engineering 5870-412-002-00 SR, Summary Report Analytical Verification of the Mechanical Stress Improvement Process (MSIP) for 12" Diameter Pipe Weldments, February 2008

EPRI Information MRP 2007-002, PWR Fleet Survey-MRP-139 Implementation Plans for Pressurizers, Rev. 2

EPRI Information MEP 2007-038, MRP-139 Interim Guidance on <4" Diameter Pipe Volumetric Exam Requirements (Mandatory Element)

EPRI Information MRP 2007-039, MRP-139 Interim Guidance on Bare Visual Exam Requirements (Mandatory Element)

Work Order (WO) 07-815000-000, Supporting and Implementing MSIP on the Unit 1 Pressurizer Six Nozzles During U1C8 RFO

Interface Review No. N2537, Pressurizer Penetration Manufacturing and Material Summary, November 22, 2004

Drawing No. CHM-2570-C-01, Watts Bar Nuclear Plant Unit 1 Pressurizer, Rev. 03

Drawing No. 1100J48, Pressurizer Outline, Rev. 9

Westinghouse Letter WT-D-11656, Transmittal of MSIP Evaluation of Misplaced Tool on Unit 1 Spray Nozzle WP-11-SE, February 26, 2008

Westinghouse Letter WT-D-11660, Transmittal of Final Element Analysis (FEA) of Misplaced Tool on Unit 1 Spray Nozzle WP-11-SE, February 29, 2008

Ultrasonic Examination (UT) Report R-1161 for Weld No. WP-10-SE, Pressurizer 15" Diameter Surge Line Dissimilar Metal, Nozzle-to-Safe End Weld, September 15, 2006 (Baseline Examination)

Ultrasonic Examination (UT) Report R-1157 for Weld No. WP-10-SE, Pressurizer 15" Diameter Surge Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 20, 2008 (Examination before MSIP Mitigation)

Ultrasonic Examination (UT) Report R-1558 for Weld No. WP-10-SE, Pressurizer 15" Diameter Surge Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 22, 2008 (Examination after Mitigation)

Ultrasonic Examination (UT) Report R-1162 for Weld No. WP-11-SE, Pressurizer 6” Diameter Spray Line Dissimilar Metal, Nozzle-to-Safe End Weld, September 16, 2006 (Baseline Examination)
Ultrasonic Examination (UT) Report R-1554 for Weld No. WP-11-SE, Pressurizer 6” Diameter Spray Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 20, 2008 (Examination before MSIP Mitigation)
Ultrasonic Examination (UT) Report R-1578 for Weld No. WP-11-SE, Pressurizer 6” Diameter Spray Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 25, 2008 (Examination afte r Mitigation)
Ultrasonic Examination (UT) Report R-1163 for Weld No. WP-12-SE, Pressurizer 8” Diameter Relief Line Dissimilar Metal, Nozzle-to-Safe End Weld, September 16, 2006 (Baseline Examination)
Ultrasonic Examination (UT) Report R-1560 for Weld No. WP-12-SE, Pressurizer 8” Diameter Relief Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 19, 2008 (Examination before MSIP Mitigation)
Ultrasonic Examination (UT) Report R-1572 for Weld No. WP-12-SE, Pressurizer 8” Diameter Relief Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 23, 2008 (Examination after Mitigation)
Ultrasonic Examination (UT) Report R-1164 for Weld No. WP-13-SE, Pressurizer 8” Diameter Safety Line Dissimilar Metal, Nozzle-to-Safe End Weld, September 16, 2006 (Baseline Examination)
Ultrasonic Examination (UT) Report R-1559 for Weld No. WP-13-SE, Pressurizer 8” Diameter Safety Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 20, 2008 (Examination before MSIP Mitigation)
Ultrasonic Examination (UT) Report R-1573 for Weld No. WP-13-SE, Pressurizer 8” Diameter Safety Line Dissimilar Metal, Nozzle-to-Safe End Weld, February 23, 2008 (Examination after Mitigation)

Section 4OA: Other: - 2515/174

Technical Manuals and Vendor Information
Cummins 2000kW Rental Package
Cummins Power Generation Generator Set Data Sheet
Cummins Power Generation Generator Set Diesel QSK60 Series Engine
Sunbelt Transformers Installation, Operation and Maintenance Manual for the Dry-Type Transformers
Insulation, Modification, and Maintenance of Insulated Cables Rated up to 15,000 Volts G-38 Rev. 20, dated 10/19/2007

Procedures
EPSIL-1, Serious Even Mitigation, Rev. 5
MA-1 Recovery from Loss of Shutdown Power and Loss of ERCW, Rev. 2
1-SI-268-1 92 Day Permanent Hydrogen Mitigation System Igniter Availability Test, Rev. 2
65KW 0556F Towable Generator, Portable Heaters, and Cabling for the IPS Freeze Protection Emergency Power Supply, Inspection and Operation, Rev. 5

Training Records
3-OT-B5b Review of B.5.b Requirements, Rev. 0 and Rosters
What is B.5.b? Presentation and Rosters
AOI-42.01 Security Events

Completed Surveillance Procedures and Test Records
WBN-1-BKR WO 07-815454-050 Breaker Maintenance, dated 3/11/2008
WBN-1-BKR WO 07-815454-051 Breaker Maintenance, dated 3/11/2008
PM 07-814106 Emergency Generator, dated 3/11/2008
PM 07-812542 Towable Diesel Generator with Trailer Kit
PM 0-FPS-026-0012 Cummins Planned Checklist, Rev. 0
DCN 51776 Recovery Requirements, Components and Connections Associated with Loss of All ERCW, dated 9/16/2004

Licensing Basis Documents
Letter – TVA to NRC, March 6, 2007, Sequoyah Nuclear Plant (SQN) Units 1 and 2, and Watts Bar Nuclear Plant (WBN) Unit 1 – Enhancement of the Capability of the Containment Hydrogen Igniters, LL44 070306 001

Problem Evaluation Reports (PERs) Generated as a Result of this Inspection
PER-145726 WBN-SIT-08-020 B.5.b Self Assessment
PER-145731 WBN-SIT-08-020 B.5.b Self Assessment
PER-145746 WBN-SIT-08-020 B.5.b Self Assessment