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

October 29, 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 05000390/2008004 AND 05000391/2008004

Dear Mr. Campbell:

On September 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 October 7, 2008, with Mr. G. Boerschig 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.

This report documents one NRC-identified finding of very low safety significance (Green) which was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC 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 05000390/2008004, 05000391/2008004
w/Attachment: Supplemental Information

cc w/encl.: See page 3

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Letter to William R. Campbell, Jr. from Eugene Guthrie dated October 29, 2008

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000390/2008004 AND 05000391/2008004

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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.:	05000390/2008004, 05000391/2008004
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Units 1 and 2
Location:	Spring City, TN 37381
Dates:	July 1 – September 30, 2008
Inspectors:	R. Monk, Senior Resident Inspector M. Pribish, Resident Inspector C. Peabody, Acting Resident Inspector
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000390/2008-004, 05000391/2008-004; 07/01/2008 - 09/30/2008; Watts Bar, Units 1 & 2; Event Followup.

The report covered a three-month period of routine inspection by resident inspectors. One NRC-identified Green finding, which is a non-cited violation (NCV), was 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: Initiating Events

- Green. A Green, NRC-identified, non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for installing flow limiting devices, on A-train residual heat removal (RHR) system valves, which were not equivalent to those described in operating procedures. The devices were installed at reduced reactor coolant system (RCS) inventory to prevent pump cavitation in the event of a loss of control air during upcoming mid-loop operation. The nonconforming device caused the RHR heat exchanger outlet valve not to fully return to its desired throttled position after verification that the flow limiting devices would limit flow within specified values. The licensee entered this issue into their corrective action program (CAP) as problem evaluation report (PER) 140284.

  The finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. The finding was evaluated by a Phase 2 analysis in accordance with the Significance Determination Process as specified by Manual Chapter 0609, Appendix G, Checklist 3 (one train of decay heat removal was degraded). The Phase 2 analysis characterized the finding as of very low safety significance (Green) because of the slow RCS heatup rate due to the low decay heat after being shutdown for 32 days and the ease of recovery from the condition. The cause of the finding was directly related to the corrective action program issue identification aspect of the cross-cutting area of Problem Identification and Resolution, in that, the licensee failed to identify the discrepancy in form, fit, and manner in which the function was accomplished between the flow limiting devices that were specified in licensee procedures and those that were actually installed (P.1(a)). (Section 4OA3.2)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near 100 percent rated thermal power (RTP) until July 24, 2008, when the B-A essential raw cooling water (ERCW) pump failed while in operation resulting in a Technical Specification (TS)-required shutdown. The unit was down-powered to approximately 77 percent RTP before an emergency TS amendment was issued, allowing the unit to return to 100 percent RTP. Full power operation continued until August 1, 2008, when a small hydrogen leak from the main generator into the stator water cooling system was detected by the licensee. The unit was down-powered incrementally by small amounts over the next four days to maintain stator bar temperatures until a decision on August 5, 2008, to shut down the unit for repairs. During the controlled reactor shutdown on August 7, 2008, the reactor was manually tripped from 48 percent RTP in response to an isolation of main feedwater caused by the failure of a low pressure feedwater heater level control valve to control feedwater heater levels. The unit was started up following repairs of the main generator hydrogen leak and returned to 100 percent RTP on August 20, 2008. Full power operation continued until the unit automatically tripped on September 20, 2008, when the main generator exciter field breaker was opened inadvertently by an operator. This resulted in a loss of main generator excitation, a main turbine generator trip and a reactor trip. The unit was returned to 100 percent RTP on September 27, 2008, and continued to operate at 100 percent RTP for the remainder of the report period.

Restart of construction on Unit 2 began in December 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

External Flood Protection Inspection

a. Inspection Scope

The inspectors performed one external flood protection measures review. 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 flood mode intersystem connections and installation procedures. The inspectors also walked down portions of the component cooling system, residual heat removal system, spent fuel pool cooling system, auxiliary feedwater system, and related ERCW and fire protection connections to verify that intersystem connection spool pieces were available to support flood mode operation for either hot shutdown or cold shutdown plant modes. The inspectors interviewed cognizant licensee personnel about site flood protection measures and plant drainage plans. The inspectors also reviewed the licensee's CAP for documents with respect to flood-related items identified in PERs written during calendar year 2007 and through September 2008. Documents reviewed are listed in 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.

- A-Train Emergency Gas Treatment System (EGTS) While B-Train EGTS Out of Service (OOS) for Corrective Maintenance
- A-Train High Head Safety Injection (HHSI) System While B-Train HHSI OOS for Preventive Maintenance
- Motor-Driven Auxiliary Feedwater (MDAFW) Trains While Turbine-Driven Auxiliary Feedwater (TDAFW) System OOS for Corrective Maintenance

b. Findings

No findings of significance were identified.

.2 Semiannual Complete System Walkdown

a. Inspection Scope

The inspectors conducted one detailed walkdown/review of the alignment and condition of the chemical volume control system-boric acid system to verify proper equipment alignment and to identify any discrepancies that could impact the function of the system and increase risk. The inspectors utilized licensee procedures, as well as licensing and design documents, when verifying that the system alignment was correct. During the walkdown, the inspectors also verified, as appropriate, that: (1) valves were correctly positioned and did not exhibit leakage that would impact the function(s) of any valve; (2) electrical power was available as required; (3) major portions of the system and components were correctly labeled, cooled, ventilated, etc.; (4) hangers and supports were correctly installed and functional; (5) essential support systems were operational; (6) ancillary equipment or debris did not interfere with system performance; (7) tagging clearances were appropriate; and, (8) valves were locked as required by the licensee's locked valve program. Pending design and equipment issues were reviewed to determine if the identified deficiencies significantly impacted the system's functions. Items included in this review were the operator workaround list, the temporary modification list, system health reports, and outstanding maintenance work requests/work orders (WOs). In addition, the inspectors reviewed the licensee's CAP to ensure that the licensee was identifying equipment alignment 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.

1R05 Fire Protection

.1 Fire Protection - Tours

a. Inspection Scope

The inspectors conducted tours of six 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.

- Auxiliary Instrument Room
- 1A-A Emergency Diesel Generator (EDG)
- 2A-A EDG
- 1B-B EDG
- 2B-B EDG
- Intake Pumping Station

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors performed one heat sink performance review. The inspectors reviewed the licensee's program for maintenance and testing of two risk-important heat exchangers in the ERCW system. Specifically, the review included the program for testing and analysis of the 'A' (1-HTX-070-0186) and 'C' (0-HTX-070-0185) component cooling system (CCS) heat exchangers. The inspectors reviewed the ERCW system description and the heat exchanger testing program document as well as completed WOs documenting the testing and visual inspection and associated corrective actions to verify that corrosion or fouling did not impact the heat exchanger from achieving its design basis heat removal capacity. The inspectors also reviewed periodic test data of ERCW and CCS flow rates as well as inlet and outlet temperatures to determine whether potential degradations were being monitored and/or prevented. Documents reviewed are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On July 1, 2008, the inspectors observed the simulator evaluations for an Operations shift crew 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 crew's 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 (EPIPs)
- 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 inspectors.

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 structures, systems, or components (SSCs) scoped into the licensee's Maintenance Rule program and to determine whether 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." The 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).

- Station Service Air Compressors a(1) Performance Improvement Plan

- Main Control Room Chiller a(1) Completed 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 assessed whether 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."

- Emergent Failure of B-Train EGTS and Auxiliary Building Gas Treatment System With the A ERCW Pump And The A Containment Spray Pump Inoperable for Maintenance
- Emergent Failure of B-A ERCW Pump While A-A ERCW Pump Was OOS for Refurbishment
- Review of Forced Outage Schedule Risk
- Emergent Inoperability of TDAFW Pump With A ERCW Pump OOS and Planned 1A EDG Maintenance
- Emergent Inoperability of B Emergency Boardroom Chiller With A ERCW Pump OOS and 1B EDG Inoperable

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed four 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 Conditions for Operation (LCOs) and the risk significance in accordance with the SDP. The inspectors evaluated whether the operability evaluations were performed in accordance with SPP-3.1, "Corrective Action Program."

- PER 148761, Oil Pressure On B Shutdown Boardroom Chiller Reading 47.5 psi; Auxiliary Unit Operator (AUO) Routine Requires >50 psi

- PER 149891, TDAFW Mini-Flow Check Valve Leakage and Suction Line Overpressure
- PER 151244, Excessive Moisture In TDAFW Pump Due to Casing Steam Leak
- PER 150546, Steam Traps Leaking Into North and South Valve Vault Rooms

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 modification did not affect system operability or availability as described by the TS and UFSAR. In addition, the inspectors determined whether: (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 08-818599-000: TACF 1-08-0007-067, Revision 1, O-XS-67-286-A , ERCW Pumps C and D Emergency Power Selector Switch; and 2-BD-211-A-A, 6.9kv Shutdown Board 2A-A, Cubicles 8 and 9

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the six 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 evaluated whether 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.” Additional documents reviewed are listed in the attachment to this report.

- WO 08-817141, Primary Makeup Water Flow Controller Replacement
- WO 08-818622-000: O-SI-67-922-A, B-A ERCW Pump Preservice Test
- WO 08-819289-000, Functional Test of Day Tank 1 Level Switches 2A-A

- WO 08-819465-000, Functional Test of ERCW to TDAFW Following Overpressure Event
- WO 08-819254, TDAFW Mini-Flow Check Valve Replacement
- WO 08-819254, Change Water Contaminated TDAFW Oil and Monitor Accumulation During Subsequent Operation

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Forced Outage Due to Hydrogen Leakage into the Stator Water Cooling System

a. Inspection Scope

The licensee began a forced outage on August 7, 2008, due to a hydrogen leak into the Unit 1 main generator stator water cooling system. The inspectors observed portions of the shutdown, maintenance activities, and startup activities to determine whether the licensee maintained defense-in-depth (DID) commensurate with the applicable TS. The inspectors monitored licensee controls over the outage activities listed below.

- Reactor Controlled Power Reduction
- Licensee Configuration Management, Including Daily Outage Reports, to Evaluate DID and Compliance With the Applicable TS When Taking Equipment OOS
- Controls Over the Status And Configuration of Redundant Safety Systems to Ensure Risk Was Minimized
- Decay Heat Removal Processes to Verify Proper Operation of Steam Generators
- 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

No findings of significance were identified.

.2 Automatic Reactor Trip Due to Human Performance Error

a. Inspection Scope

The licensee began a forced outage on September 20, 2008, following a human performance error resulting in the automatic trip of the unit. The inspectors observed maintenance activities, and startup activities to determine whether the licensee maintained DID commensurate with the applicable TS. The inspectors monitored licensee controls over the outage activities listed below.

- Review of Plant Response to the Reactor Trip

- Licensee Configuration Management, Including Daily Outage Reports, To Evaluate DID and Compliance With the Applicable TS When Taking Equipment OOS
- Controls Over the Status and Configuration of Redundant Safety Systems to Ensure Risk Was Minimized
- Decay Heat Removal Processes to Verify Proper Operation of Steam Generators
- 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
- 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

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed three surveillance tests and/or reviewed test data of selected risk-significant SSCs, listed below, to assess, as appropriate, whether the testing met: (1) TS requirements; (2) the UFSAR; (3) SPP-8.0, "Testing Programs;" (4) SPP-8.2, "Surveillance Test Program;" and, (5) 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 Test:

- 0-SI-82-12-A, Monthly Start And Load Test 2A-A EDG (WO 08-116940-000)

In-Service Tests:

- 0-FOR-70-2, 2B-B CCS Pump Quarterly Performance Test (WO 08-811727-000)
- 1B-B HHSI Pump Quarterly Performance Test (WO 08-819207-000)

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 to verify that the emergency response organization was properly classifying the event in accordance with EPIP-1, "Emergency Plan Classification Flowchart," and making accurate and timely notifications and protective action recommendations in accordance with EPIP-3, "Alert," EPIP-4, "Site Area Emergency," and the Radiological Emergency Plan. In addition, the inspectors evaluated whether 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: failed fuel and steam generator tube rupture leads to site area emergency

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verifications

The inspectors sampled licensee submittals for the five PIs listed below. To verify the accuracy of the PI data reported during the periods listed, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline, Revision," were used to verify the basis in reporting for each data element. Additional documents reviewed are listed in the attachment.

Mitigating Systems Cornerstone PI

- Mitigating System Performance Index (MSPI) - HHSI
- MSPI - Cooling Water
- MSPI - Heat removal
- MSPI - RHR
- MSPI - Emergency AC

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 examining daily PER summary reports and attending a sampling of daily PER review meetings.

4OA3 Event Followup

.1 Unit 1 Reactor Trip – August 7, 2008

a. Inspection Scope

The inspectors responded to a Unit 1 manual reactor trip that occurred on August 7, 2008. The inspectors discussed the preliminary cause of the scram with licensee management, operations, and engineering. The inspectors reviewed unit parameters and system response to verify whether equipment responded to the scram as designed. The inspectors also reviewed parts of the licensee's post-scram review. The inspectors reviewed the initial licensee event notification to determine if it met regulatory requirements.

b. Findings

No findings of significance were identified.

.2 Unit 1 Reactor Trip – September 20, 2008

a. Inspection Scope

The inspectors responded to a Unit 1 automatic scram that occurred on September 20, 2008. The inspectors discussed the preliminary cause of the scram with licensee management, operations, and engineering. The inspectors reviewed unit parameters and system response to verify whether equipment responded to the scram as designed. The inspectors also reviewed parts of the licensee's post-scram review report and discussed the initial preliminary root cause with the licensee's incident investigation team. The inspectors reviewed the initial licensee event notification to determine if it met regulatory requirements.

b. Findings

No findings of significance were identified.

.3 (Closed) Unresolved Item (URI) 05000390/2008002-03, Failure to Ensure Adequate Design and Testing of RHR Flow Limiters

a. Inspection Scope

The inspectors completed a review and characterization of URI 50-390/2008002-03, Failure to Ensure Adequate Design and Testing of RHR Flow Limiters, and evaluated the risk significance of the issue.

b. Findings

Introduction: A Green, NRC-identified NCV of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified for installing flow limiting devices which were not similar to those described in procedure GO-10, "Reactor Coolant System Drain and Fill Operations," Appendix Q, revision 37. This resulted in one of the RHR heat exchanger flow control valves not fully returning to its desired throttled position after verification that the flow limiting devices would limit flow within specified values. The licensee entered this issue into their CAP as PER 140284.

Description: On March 14, 2008, inspectors observed the licensee preparing the RHR system for mid-loop operation. Procedure GO-10 Appendix Q, required flow limiting devices to be installed on both the RHR Heat Exchanger Bypass Valve, 1-FCV-74-32 and RHR Heat Exchanger Outlet Valve, 1-FCV-74-16 for A train. The purpose of these devices was to limit RHR flow in the event of a loss of nonsafety-related instrument air which would result in the valves failing open thus causing pump cavitations and potential air binding of the RHR pumps during mid-loop operations. Because field personnel installing the devices experienced difficulty configuring the devices, the inspectors questioned what verification was to be performed to verify that the devices would limit flow as required. Although the licensee had not planned to verify properly device operation, the licensee decided to perform one. The flow limiter on the bypass valve was successfully verified. However, during the outlet valve verification, the valve while being stroked to its desired throttled position, became bound. Once the flow limiter was removed the valve stroked properly. The devices were subsequently reinstalled and the licensee proceeded into mid-loop.

Appendix Q step one states “FABRICATE blocking devices suitable for application to the operators of the applicable valves on the operating train (similar to Figure 1).” The inspectors observed that the flow limiting devices installed were not similar to Figure 1 in terms of fit and form, and the function to limit flow was accomplished in a somewhat different fashion as described below. Figure 1 shows a two-inch outside diameter cylinder with a 7/8 inch axial hole and three holes on each side drilled into the device for six bolts, with length of the device to be determined in the field. Due to wording in Appendix Q, which directs the installer to have flow raised to 2300 gpm prior to installation, the intent was to clamp the device onto the valve stem at a point where it was up against the valve operator inhibiting further outward (open) motion. The actual devices observed, appeared to be a length of tubing split axially, and held on the actuator stem with a single center mounted hose clamp. The auxiliary operator had difficulty installing the devices due to the use of rubber gloves and the wall of the tubing was thin and difficult to hold aligned while the hose clamp was tightened. Improperly installed devices would slide on the valve stem as opposed to clamping as shown in the procedure figure and would therefore use their length as a valve motion stop setting.

Analysis: Installing flow limiting devices on RHR flow control valves which were not similar in form, fit and manner of function to the device shown on procedure GO-10, Appendix Q, Figure 1 was a performance deficiency. The finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown because the flow limiting device prevented the outlet valve from being fully reopened to its desired throttled position. Installation of the mis-fabricated device slightly reduced the capability of the RHR systems to provide core heat removal. The inspectors determined, using Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determination Process,” Checklist 3, that a Phase 2 analysis was required since the finding represented degradation of one train of decay heat removal. A regional Senior Reactor Analyst performed the Phase 2 analysis and concluded that the finding was characterized as of very low safety significance (Green). The performance deficiency was a precursor finding, no actual loss of RHR occurred. The major assumptions and influential factors in the analysis included that the plant was in Mode 5 at 32 days after shutdown with low decay heat; all preparations for going to midloop in place; and the unit was in plant operating state 2 with the RCS vented and RCS temperature at 74 degrees Fahrenheit. The initiating event likelihood was determined to be 1.0E-3 due to the actual affect of the installed flow limiting devices, the low decay heat, the fact that RCS temperature was not regulated with RCS flow changes, and the ease of recovery when one limiter restricted flow slightly lower than desired when tested which would result in a slow RCS heatup. The exposure period was approximately 32 hours. The dominant accident sequence was a loss of flow control (LOI initiator) and failure to establish RCS injection prior to core damage. Due to the timing of the performance deficiency (at 32 days after shutdown) the large early release frequency impact from a potential loss of RHR at midloop would not be significant. The resultant core damage frequency was < 1E-6 per year, and the external events contribution was not sufficient to increase the risk above this threshold.

The cause of the finding was directly related to the corrective action program issue identification aspect of the cross-cutting area of Problem Identification and Resolution, in that, the licensee failed to identify the discrepancy in form, fit, and manner in which the function was accomplished between the flow limiting devices that were specified in procedure GO-10, Appendix Q, and those that were actually installed (P.1(a)).

Enforcement: 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, on March 14, 2008, flow limiting device installation was not accomplished in accordance with procedure GO-10, in that, the form, fit and manner of function of the installed flow limiting devices were different from the flow limiting device shown on procedure GO-10, Appendix Q, Figure 1. Because this violation was of very low safety significance and it was entered into the licensee's CAP as PER 140284, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 5000390/2008004-01, Installation of Different RHR Flow Limiters Than Prescribed by Procedure.

4OA5 Other Activities

.1 (Open) Temporary Instruction (TI) 2515/176, Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing

a. Inspection Scope

The objective of this TI was to gather information to assess the adequacy of nuclear power plant EDG endurance and margin testing as prescribed by plant-specific TS. The inspectors interfaced with the appropriate station staff to obtain the information specified in Attachment 1 of the TI Worksheet. The TI applies to all operating nuclear power reactor licensees that use EDGs as the onsite standby power supply. The inspectors verified the accuracy of the information by review of TS, EDG Design Basis Event loading calculations, EDG endurance run test procedures, test data from the last three endurance tests performed on each EDG, EDG ratings, and EDG operating history. The information gathered will be forwarded to Nuclear Reactor Regulation/Division of Engineering/Electrical Engineering Branch (NRR/DE/EEEB) for further review to assess the adequacy and consistency of EDG testing at nuclear stations.

b. Findings and Observations

The TI is presently scheduled to be open until August 31, 2009, pending completion of the NRR/DE/EEEB review.

.2 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 reviews and inspection activities.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

The inspectors presented the inspection results to Mr. G. Boerschig and other members of licensee management at the conclusion of the inspection on October 7, 2008. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

L. Belvin, Radiation Protection Manager
M. Brandon, Licensing and Industry Affairs Manager
B. Hunt, Operations Superintendent
G. Mauldin, Site Engineering Manager
F. Leonard, Engineering
G. Boerschig, Plant Manager
M. McFadden, Site Nuclear Assurance Manager
A. Scales, Operations Manager
M. Skaggs, Site Vice President
S. Smith, Operations Superintendent
U. Totora, Site Support Manager
D. Voeller, Maintenance and Modifications Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

TI 2515/176	TI	Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing (Section 4OA5)

Opened and Closed

5000390/2008004-01	NCV	Installation of Different RHR Flow Limiters Than Prescribed by Procedure (Section 4OA3.2)

Closed

50-390/2008002-03	URI	Failure to Ensure Adequate Design and Testing of RHR Flow Limiters (Section 4OA3.2)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather

Updated Final Safety Analysis Report (UFSAR) Sections 2.4.14, 3.4
AOI-7.01, Maximum Probable Flood, Revision 16
AOI-7.07, Shutdown of CCS and Changeover to ERCW, Rev 2
Technical Requirement Instruction, 0-TRI-100-1, Flood Protections Communications, Rev. 6
Maintenance Instruction, MI-17.018, Flood Preparation-High Pressure Fire Protection System Spool Pieces, Rev. 10
Maintenance Instruction, MI-17.019, Flood Preparation-Auxiliary Charging System Spool Piece, Rev. 6
Maintenance Instruction, MI-17.020, Flood Preparation-Sample Heat Exchanger Spool Pieces, Rev. 5
Maintenance Instruction, MI-17.021, Installation of Spool Pieces Between SFPC System and RHR System, Rev. 6
Maintenance Instruction, MI-17.022, Flood Preparation-Sample Heat Exchanger Spool Pieces, Rev. 5
Maintenance Instruction, MI-17.023, Flood Preparation-Reactor Coolant Spool Pieces, Rev. 5
Maintenance Instruction, MI-17.024, Flood Preparation-Ice Condenser Spool Piece, Rev. 5
Individual Plant Examination, Watts Bar Internal Flood Analysis Section E.1.5.2.1, Intake Pumping Station, Revision 0
Individual Plant Examination for External Events – High Winds, Floods, and other External Events, Attachment 5 Section 5.6
WB-DC-20-28, Intake Pumping Station Watertight Doors at Elevation 722.0, Revision 4
WB-DC-20-31, Plant Drainage, Revision 3
WB-DC-40-64, Design Basis Events Design Criteria, Section 4.4-Design Basis Flood, Revision 11
WB-DC-40-29, Flood Protection Provisions, Section 4.11 and Table 4.1-2, Revision 9
PER 120918, Plant response to IN 2007-01, Operating Experience Concerning Hydrostatic Barriers
PER 137274, Flood Mode Charging Pump determined to be in Alert range on flow and vibration
PER 139762, Licensee discovered inappropriate gasket material stage for one flood mode spool piece
PER 146803, Licensee discovered spool piece pre-staged installation tools missing.
PER 152172, Margin Management issue for non-conservative input data to Probable Maximum Flood calculation
PER 152177, Margin Management issue related to changes in River Service Operations operating policy for reservoirs as described in the FSAR.
WO 07-812334 Replacement of 20 " piping supply to flood mode spool piece.
WO 07-813047 Performance of TI-50.048, Flood Mode Auxiliary Charging Pump 1A Performance Test.
WO 07-813073 Performance of TI-50.049, Flood Mode Auxiliary Charging Pump 1B Performance Test.
WO 07-813071 Performance of TI-50.046, Flood Mode Auxiliary Charging Booster Pump 1A Performance Test.
WO 07-813051 Performance of TI-50.047, Flood Mode Auxiliary Charging Booster Pump 1B Performance Test.

Section 1R04: Equipment Alignment

Drawings:
1-47W811-1, Flow Diagram, Safety Injection System, Rev. 46
1-47W809-2, Flow diagram, Chemical and Volume Control System (Boron Recovery), Rev. 21
1-47W809-5, Flow Diagram, Chemical and Volume Control System (Boric Acid), Rev. 22
Work Order 08-816526-000, Annunciator window, 135F, “B” Boric Acid Tank Level Indicator Failing Low, 06-17-2008
Work Order 08-817141-000, Recorder connected to 1-LPF-62-143 Primary Water Flow to Blender, 07/05/2008

Section 1R07: Heat Sink Performance

Procedures:
TI-79.000, Generic Letter 89-13 Heat Exchanger Test Program, Rev. 0010

System Descriptions:
N3-67-4002, Essential Raw Cooling Water System, Rev. 0022

Work Orders:
WO 02-013209-000, CCS HX C Eddy Current Test, 9/23/2003
WO 03-021246-000, CCS HX A Eddy Current Test, 3/5/2005
WO 05-815458-000, Disassemble, Clean, and ECT CCS HX C, 9/28/2006
WO 05-819545-000, Disassemble, Clean and Inspect CCS HX A, 10/25/2006
WO 05-820843-000, CCS HX C Performance Test, 12/19/2006
WO 05-820844-000, CCS HX C ERCW Side Flow and DP Monitoring, 8/15/2006
WO 07-811157-000, Disassemble, Clean, and ECT CCS HX C, 3/3/2008
WO 07-811067-000, Disassemble, Clean, and Inspect CCS HX A, 2/25/2008
WO 07-816398-000, CCS HX A Performance Test, 4/11/2008

Section 1R19: Post-Maintenance Testing

Completed Test Procedures:
2-LS-018-0062A/3-A, Functional Test of Day Tank 1 Level Switches Bay 2A-A, completed 8/11/2008, WO 08-819289-000
0-SI-67-922-A, Essential Raw Cooling Water Pump B-A Preservice Test, Rev. 0, completed 7/25/2008, WO 08-818622-000

Work Orders:
WO 08-819465-000, Functional test of ERCW to TDAFW following over-pressurization event, 8/14/2008

Section 4OA1: Performance Indicator Verifications
MSPI Derivation Reports of Unavailability and Unreliability for HHSI, Cooling Water, Heat removal, RHR and Emergency AC systems

Section 4OA2: Identification and Resolution of Problems

PERs written as a result of this report
PER 149257, Water is leaking from the ceiling near 2B-1B RX MOV board
PER 151046, No protected equipment sign on the door of the 1A SIP room, although the 1B SIP

was OOS and 1A was protected
PER 148243, Inadequate instructions for replacing and installation of the controller and no PMT specified for Primary Water Blender Flow Control
PER 151026, AUO did not have keys to spaces required for EOP actions in a timely manner
PER 151252, CAP process failed to recognize an operability issue (ODCM TS 2.0.3 not met) when Steam Generator Blowdown effluent release valve (1-FCV 15-44) was found out of surveillance grace
PER 151962, PER Screening Committee composition not IAW PIDP-4
PER 152038, Insulation missing from 1B RHR Hx
PER 152109, Missed Unplanned Transient input
PER 152229, Missed MSPI failure input of 1B CCP
PER 152372, CAP process failed to recognize an operability issue when RM-106 found inop during calibration

Section 4OA5: Other Activities

Licensing Documents:
Technical Specifications, Unit 1, Amendment 39

Procedures:
0-SI-82-13, 24 Hour Load Run – 1A-A
0-SI-82-14, 24 Hour Load Run – 1B-B
0-SI-82-15, 24 Hour Load Run – 2A-A
0-SI-82-15, 24 Hour Load Run – 2B-B

Calculations:
WBN-EEB-MS-TI03-0012, Diesel Generator Loading Analysis

Other:
WO 07-821115-000, 0-SI-82-13, 24 Hour Load Run – 1A-A, 8/1/08
WO 06-813039-000, 0-SI-82-14, 24 Hour Load Run – 1B-B, 1/3/07
WO 07-821234-000, 0-SI-82-15, 24 Hour Load Run – 2A-A, 8/7/08
WO 07-821077-000, 0-SI-82-15, 24 Hour Load Run – 2B-B, 8/20/08