UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

February 13, 2013

Mr. Joseph W. Shea
Vice President Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3D-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000390/2012005

Dear Mr. Shea:

On December 31, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Watts Bar Nuclear Plant, Unit 1. The enclosed inspection report documents the inspection results which were discussed on January 11, 2012, with Mr. T. Cleary and other members of the Watts Bar 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.

One NRC-identified finding and two self-revealing findings of very low safety significance (Green) were identified during this inspection. Additionally, two unresolved items were opened. These findings were determined to involve violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Watts Bar Nuclear Plant.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II; and the NRC Resident Inspector at the Watts Bar Nuclear Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (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/

Scott M. Shaeffer, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos.: 50-390
License No.: NPF-90

Enclosure: NRC Inspection Report 05000390/2012005
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Letter to Joseph Shea from Scott Shaeffer dated February 13, 2013

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000390/2012005

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-390
License No.:	NPF-90
Report No.:	05000390/2012005
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Unit 1
Location:	Spring City, TN 37381
Dates:	October 1 through December 31, 2012
Inspectors:	R. Monk, Senior Resident Inspector K. Miller, Resident Inspector M. Speck, Senior Emergency Preparedness Inspector, RII (Section 1EP5, 1EP6) R. Lanyi, Operations Engineer, RII (Section 1R11) D. Failla, Construction Inspector, RII (Section 1R20) J. Hamman, Project Engineer, RII (Section 1R20) E. Patterson, Resident Inspector, RII (Section 1R20, 1EP6) R. Baldwin, Senior Operations Engineer, RII (Section 1R11)
Approved by:	Scott M. Shaeffer, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000390/2012-005; 10/01/2012 – 12/31/2012; Watts Bar, Unit 1; Adverse Weather Protection, and Identification and Resolution of Problems.

The report covered a three-month period of inspection by resident inspectors and announced inspections by regional inspectors. Three Green findings were identified which involved non-cited violations (NCVs) of NRC requirements. Also, two unresolved items were opened. The significance of most findings is identified by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP); the cross-cutting aspect was determined using IMC 0310, "Components Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process" Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Emergency Preparedness

- Green: A self-revealing non-cited violation (NCV) of 10 *Code of Federal Regulations* (CFR) 50.54q(2) for failure to follow the approved emergency plan. Specifically, on August 10, 2012, state officials were not notified within 15 minutes of the declaration of an Unusual Event. State notification is a risk-significant planning standard requirement required by 10 CFR 50.47(b)(5), 10 CFR 50 Appendix E, Section IV.D.3 and Section 5.2.1, of the licensee's Radiological Emergency Plan.

  The issue was greater than minor because it was associated with the Emergency Planning cornerstone attribute of Emergency Response Organization performance during an actual event. The finding affected the cornerstone objective in that timely notification is critical to ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The inspectors reviewed this finding using IMC 0609, Appendix B, Emergency Preparedness Significance Determination Process, Attachment 1, Failure to Implement (Actual Event) Significance Logic. The finding was determined to be of very low safety significance because it was a failure to implement during an Unusual Event. The finding had a cross-cutting aspect in the area of Human Performance, Decision-Making, because the unit supervisor, in the absence of the shift manager, did not effectively fulfill his responsibility to direct or perform required state communications within the required 15 minute time period as required by the Radiological Emergency Plan. (H.1(a). (Section 1EP5)

Cornerstone: Mitigating Systems

- Green. A NCV of 10 CFR 50 Appendix B, Criterion III, Design Control, for the licensee's failure to adequately develop and implement ice condenser ice basket repairs in accordance with approved engineering and maintenance documents. Specifically, the inspectors observed that repairs to six damaged ice condenser ice baskets, previously signed off as complete in the work order (WO) by the installers and following Quality Control inspection and acceptance were not in accordance with the design and maintenance WO documents. The licensee initiated Problem Evaluation Reports (PERs) 623040 and 626983 to address the inspector-identified deficiencies.

The licensee's failure to adequately develop and implement ice condenser ice basket repairs in accordance with approved engineering and maintenance documents was a performance deficiency. The inspectors reviewed Inspection Manual Chapter (IMC) 0612 and determined that the finding was more than minor because the deficiencies were not identified by the licensee and would have remained unidentified at least for the duration of the upcoming fuel cycle. Without the specified repairs being properly implemented on the damaged ice baskets, there was no reasonable assurance they were capable of performing their design function, and there was also potential for damage to adjacent ice baskets obstructing open flow paths, in the event the ice condenser was required to perform its design function. Using the Initial Characterization of Findings guidance of IMC 0609, the inspectors determined that the finding was of very low safety significance (Green) because no actual loss of safety function occurred. The cause of the finding had a cross-cutting aspect in the area of effective supervisory/management oversight in the Work Practices component. It was directly related to the licensee not ensuring adequate supervisory and management oversight of work activities, including the licensee engineering personnel that prepared and reviewed the ECP, the contractors that performed the repair work and the Quality Control personnel that performed the repair inspection and acceptance. (H.4 (c)). (See Section 1R18)

- Green. A self-revealing NCV of Technical Specifications (TS) 5.7.1, Procedures, was identified for failing to adhere to OPDP-1, Conduct of Operations, Section 5.1, Procedure Adherence. The licensee failed to ensure a jumper was removed prior to placing the steam generator blowdown system into service per System Operating Instruction 90.01, Rev. 29, Liquid Process Radiation Monitors, step 5.5 [10]. This was a performance deficiency and a finding. The finding was more than minor because, if left uncorrected, it could lead to a more significant safety issue, a radioactive release, and was associated with the Mitigating Systems Cornerstone attribute of equipment performance (reliability) and adversely affected the cornerstone objective. The finding was evaluated using the SDP Phase I and was determined to be a finding of very low safety significance because actual high contamination levels did not occur within the steam generators during the period that the jumper was installed. The licensee entered this issue into the corrective action program as PER 637279. The finding directly involved the cross-cutting area of Human Performance under the procedural compliance aspect of the work practices component; in that the procedural requirements of System Operating Instruction 90.01 were not met. (H.4(b)) (Section 4OA3)

B. Licensee-Identified Violations

- Technical Requirements (TR) 3.4.2, Pressurizer Temperature Limits, required that pressurizer heatup rate be limited to ≤100° F in any 1-hour period. The TR action statement A.1 required that the limit be restored within 30 minutes. Contrary to the above, at or around 0834, October 10, 2012, while Unit 1 was in Mode 5, the licensee determined that the pressurizer vapor space heatup rate limit of ≤100° F in any 1-hour period had been exceeded. (Section 4OA7)

- 10 CFR 50 Appendix B, Criterion III, Design Control, states in part that measures shall be established to assure that regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the licensee failed to control both ASME III code and non-ASME code materials during relocation of WBN-1-PDT-030-0042-G, WBN-1-PDT-030-0045-D, and WBN-1-PDT-030-0043-F installed by Design Change Notice 58382 stages 8, 10, and 11. Portions of the installed material did not meet American Society of Mechanical Engineers (ASME) Section III Class 2, TVA Class B design requirements resulting in non-ASME code material being used in the fabrication of ASME code components. (Section 4OA7)

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period in a refueling outage. The unit was returned to full power operation on November 3, 2012. The unit continued to operate at or near 100 percent rated thermal power (RTP) until December 30, 2012, when it was ramped down to 20 percent power to repair the pneumatic controllers on #2 and #3 steam generator feedwater regulating valves. The unit was ramping back to 100 percent RTP at the end of the reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

Readiness for Seasonal Extreme Weather Readiness

a. Inspection Scope

The inspectors reviewed licensee actions taken in preparation for low temperature weather conditions to limit the risk of freeze-related initiating events and to adequately protect mitigating systems from its effects. The inspectors reviewed licensee procedure 1-PI-OPS-1-FP, Freeze Protection, and walked down selected components associated with the four areas listed below to evaluate implementation of plant freeze protection, including the material condition of insulation, heat trace elements, and temporary heated enclosures. Corrective actions for items identified in relevant problem evaluation reports (PERs) and work orders (WOs) were assessed for effectiveness and timeliness. This inspection satisfied one inspection sample for extreme weather readiness. Documents reviewed are listed in the Attachment.

- Performed walkdowns for extreme weather preparations at the intake pumping station, the refueling water storage tank, feedwater flow transmitters, and diesel building

b Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial System 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. This includes also that redundant trains are returned to service properly. The inspectors reviewed the functional system descriptions, Updated Final Safety Analysis Report (UFSAR), system operating procedures, and technical specifications (TS) to determine correct system lineups for the current plant conditions. The inspectors performed walkdowns of the systems to verify that critical components were properly aligned and to identify any discrepancies which could affect operability of the redundant train or backup system. Documents reviewed are listed in the Attachment.

- 1B containment spray (CS) pump
- 1B auxiliary feedwater (AFW) pump
- 1B RHR pump

b. Findings

No findings were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors conducted one detailed walkdown/review of the alignment and condition of the main steam supply to the turbine-driven auxiliary feedwater (TDAFW) pump 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, to verify 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 and work orders (WOs). In addition, the inspectors reviewed the licensee's corrective action program (CAP) to ensure that the licensee was identifying equipment alignment problems and to ensure they were properly addressed for resolution. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Fire Protection Tours

a. Inspection Scope

The inspectors conducted tours of the 11 areas important to reactor safety, listed below, to verify the licensee's implementation of fire protection requirements as described in the Fire Protection Program, Nuclear Power Group Standard Programs and Processes (NPG-SPP)-18.4.6, Control of Fire Protection Impairments, NPG-SPP-18.4.7, Control of Transient Combustibles, NPG-SPP-18.4.8, 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. This activity constituted 11 inspection samples.

- Cable spreading room
- 480 V reactor (RX) motor-operated valve (MOV) board room 1A
- 480 V RX MOV board room 1B
- 480 V RX MOV board room 2A
- 480 VRX MOV board room 2B
- Vital battery room I, II, III, IV, V
- Auxiliary instrument room

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed internal flood protection measures for the control building. Flood protection features were examined to verify that they were installed and maintained consistent with the plant design basis. The inspectors also reviewed the licensee flooding study calculation for determining maximum flood level in the turbine building for condenser circulating water (CCW) failures that could impact the control building and confirmed that flood mitigation features such as drains and curbs were not degraded in such a manner as to adversely impact the conclusions of the study. Documents reviewed are listed in the Attachment. This activity constituted one inspection sample.

b. <u>Findings and Observations</u>

<u>Introduction</u>: Engineering justification for the design of floor hatches (WBN-1-EQH-271-0008 and WBN-2-EQH-271-0008) in the 708.0' elevation (El.) of the control/turbine building.

<u>Description</u>: The inspectors reviewed Final Safety Analysis Report (FSAR), Section 3.8.4.1.1, which requires the two equipment hatches at elevation 708.0' in the control building to be watertight. Design Criteria WB-DC-20-21, Miscellaneous Steel Components for Category I Structures, Revision 13, covers the requirements for the equipment and, according to TVA Drawing 48N1306, the hatches are designed to be watertight. Drawing 48N1306 notes: "9. Seals and gaskets by Heavy Equipment Group," and "13. Gaskets shall be affixed to cover plates with waterproof cement or equal to assure that they remain attached to covers when covers removed." There is no information on the drawing regarding the material specifications of the gaskets, only the thickness: "1/8" gasket." Design Criteria Document WB-DC-40-60, Special Hatches and Manways, Revision 6, Section 3.12.2.2, states that the hatches must withstand a pressure of 1.3 psi from topside (water to El. 711.0' due to a turbine building flood resulting from a rupture in the CCW system). Service Request (SR) 427917 (initiated September 5, 2011) reported "Water leaking down into the EBR Chiller Room – Water leaking through the equipment hatch (WBN-2-EQH-271-0008) seals and dripping down into the EBR Chiller room and pooling in front of the door. Repair/replace leaking hatch seals and the seals around the coffer dam. The source of water appears to be from rain water entering the TB around the Steam & Feed line penetrations in the NE corner elevation 729' Unit-2 side. Cover/close/or seal these penetrations to prevent rain water from entering TB." This SR was closed to WO 112678945 and the WO was cancelled on May 9, 2012. It is unlikely that the hatches are capable of being watertight at a pressure of 1.3 psi from topside (water to El. 711.0') if they leak rain water from the 708.0' El. floor.

Calculation WCG-1-1591, Seismic Evaluation of Watertight Equipment Hatches in Control Building at El. 708.0, provides some evaluation of structural elements of the hatches. This calculation does not provide any evaluation of the shear stress in the connecting screws due to hatch deflection/elastic deformation nor does the calculation evaluate the design adequacy of the mounting frame attached to the concrete opening under a live load (22,500 lbs.). Pending additional information from the licensee which can verify that there was adequate engineering justification for the design of floor hatches (WBN-1-EQH-271-0008 and WBN-2-EQH-271-0008) in the 708.0' El. of the control/turbine building, this item is identified as unresolved item (URI) 050000390/2012005-01, Engineering Justification for Design of Control Building Watertight Hatches.

1R11 Licensed Operator Requalification

.1 Quarterly Review

a. Inspection Scope

On October 31, 2012, the inspectors observed a simulator evaluation for Pilot Crew B, 3-OT-SR E0009, False SI/ATWS/SGTR. The plant conditions led to a Site Area Emergency level classification. Performance Indicator credit was taken.

On November 7, 2012, the inspectors observed a simulator evaluation for Crew 1A, 3-OT-SRE 0021B, Feedwater break/loss of 6.9KV SDBD1B/loss of secondary heat sink. The plant conditions led to a Site Area Emergency level classification. Performance Indicator credit was taken.

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 AOIs, and emergency operating instructions
- 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.

Inspectors observed and assessed licensed operator performance in the plant and main control room, particularly during periods of heightened activity or risk and where the activities could affect plant safety. Inspectors reviewed various licensee policies and procedures such as OPDP-1, Conduct of Operations, NPG-SPP-10.0, Plant Operations and GO-4, Normal Power Operation.

Inspectors utilized activities such as post maintenance testing, surveillance testing and refueling and other outage activities to focus on the following conduct of operations as appropriate;

- Operator compliance and use of procedures.
- Control board manipulations.
- Communication between crew members.
- Use and interpretation of plant instruments, indications and alarms.
- Use of human error prevention techniques.
- Documentation of activities, including initials and sign-offs in procedures.

- Supervision of activities, including risk and reactivity management.
- Pre-job briefs.

b. Findings

No findings were identified.

.2 Biannual Review

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of November 13, 2012, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the facility licensee in implementing requalification requirements identifies in 10 CFR Part 55, "Operators' Licenses." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure 71111.11, "Licensed Operator Requalification Program." The inspectors also evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations using ANSI/ASN-3.5-1985, "American National Standard for Nuclear Power Plant Simulators For Use In Operator Training and Examination." The inspectors observed two crews during the performance of the operating tests. Documentation reviewed included written examinations, Job Performance Measures (JPMs), simulator scenarios, licensee procedures, on-shift records, simulator modification request records, simulator performance test records, operator feedback records, licensed operator qualification records, remediation plans, watchstanding records, and medical records. The records were inspected using the criteria listed in Inspection Procedure 7111.11. Document reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.3 Annual Review

a. Inspection Scope

On December 7, 2012, the licensee completed the annual requalification operating examinations required to be administered to all licensed operators in accordance with 10 CFR 55.59(a)(2). The inspectors performed an in-office review of the overall pass/fail results of the individual operating examinations and the crew simulator operating examinations in accordance with Inspection Procedure (IP) 71111.11, "Licensed Operator Requalification Program." These results were compared to the thresholds established in Inspection Manual Chapter (IMC) 0609, "Significance Determination

Process," Appendix I, "Operator Requalification Human Performance Significance Determination Process."

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the three performance-based problems listed below. A review was performed 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 NPG-SPP-03.4, 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).

- Review of basis to reclassify main steam safety valves from category a(1) to category a(2)
- Review of basis to reclassify the turbine control system from category a(1) to category a(2)
- Review of a(1) performance improvement plan for the auxiliary control air system

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated, as appropriate, for the three 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); NPG-SPP-07.0, Work Control and Outage Management; NPG-SPP-07.1, On Line Work Management; and TI-124, Equipment to Plant Risk Matrix. This inspection satisfied five inspection samples for Maintenance Risk Assessment and Emergent Work Control.

- Risk assessment of work week 608 with 1B residual heat removal (RHR) pump out of service (OOS) for surveillance and D-A essential raw cooling water (ERCW) pump OOS for motor replacement
- Emergent risk assessment for the failure of #3 main steam loop pressure, 1-PT-1-20A while D-A ERCW pump OOS for motor replacement
- Risk assessment for work week 612 with 1B safety injection pump and D-A ERCW pump OOS for routine maintenance and testing of the TDAFW pump

b. Findings

No findings 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 Conditions for Operation (LCOs) and the risk significance in accordance with the significant determination process (SDP). The inspectors verified that the operability evaluations were performed in accordance with NPG-SPP-03.1, Corrective Action Program.

- Prompt determination of operability (PDO) for PER 606039, Failure of C-A ERCW pump to load shed
- Past operability review for PER 637674 for containment isolation valve 1-FCV-61-193
- PDO for PER 630120 for error in TDAFW pump speed indication

- PDO for PER 626307 for reverse leak-by on SGBD # 4 containment isolation valve 1-FCV-1-0032A
- Functional evaluation (FE) for PER 652770, Turbine building flooding could exceed CLB flood El. 711 ft due to lack of 1E electrical supply to supplemental condenser circulating water

b. Findings

No findings were identified.

1R18 Plant Modifications

a. Inspection Scope

The inspectors reviewed one permanent plant modification against the requirements of NPG-SPP-09.3, Plant Modifications and Engineering Change Control, and NPG-SPP-09.4, 10 CFR 50.59 Evaluation of Changes, Tests, and Experiments, and verified that the modification did not affect system operability or availability as described by the TS or the UFSAR. In addition, the inspectors determined whether: (1) the installation of the permanent 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.

- Engineering Document Change (EDC) E-50607, Revision A, Provide an Optional Lower Ice Basket Support for Damaged Ice Baskets
- Equivalent Change (EQV) 60275, Revision A, Provide Alternate Support Mechanism for Damaged Ice Baskets

b. Findings

.1 Introduction: A Green, NRC-identified NCV of 10 CFR 50 Appendix B, Criterion III, Design Control, was identified for the licensee's failure to adequately develop and implement ice condenser ice basket repairs in accordance with approved engineering and maintenance documents.

Description: On October 11, 2012, with the plant in Mode 5, Cold Shutdown, the inspectors reviewed outage maintenance WO 113393057 which specified the installation of new hardware on a total of six ice baskets that had been damaged, apparently due to ice condenser maintenance. Engineering had provided guidance for the repairs via two engineering documents: EDC E-50607, Revision A, Provide an Optional Lower Ice Basket Support for Damaged Ice baskets, and EQV 60275, Revision A, Provide Alternate Support Mechanism for Damaged Ice Baskets. On October 11, 2012, the inspectors performed a field observation of the repairs, previously signed off as complete in the WO by the installers and following Quality Control inspection and acceptance on October 10, 2012. The inspector's field observation of the signed-off physical work identified a number of as-left repair conditions that were not in accordance with the design and maintenance WO documents. The inspectors noted the following deficiencies: 1) the lower ice basket supports that were installed the day before were not engaged on the ligaments at the bottom of the ice baskets, since the lower bracket groves were not designed wide enough to accommodate the basket end assembly material thickness. 2) a set-screw on one of the recently installed lower brackets was not tightened, as-designed. 3) less than the minimum number of wire ropes had been installed on damaged basket D7 in Bay 2 for the extent of damaged ligaments; there apparently was a need for at least three wire ropes per engineering design documents but only two wire ropes were installed. The licensee initiated PER 623040 to address the inspector-identified deficiencies. Engineering proceeded to redesign the lower ice basket supports (EDC E-50607) which involved machining a wider groove in the lower bracket. Repair work on the subject ice baskets was re-performed on October 13, 2012, to resolve the inspector-identified deficiencies. On October 17, 2012, the inspectors performed a field observation of the repairs, previously signed off as complete in the WO by the installers and following Quality Control inspection and acceptance on October 13, 2012. The inspectors noted the following deficiency: a set-screw on one of the recently installed lower brackets on damaged basket D5 in Bay 23 was not tightened, as-designed. The licensee initiated PER 626983 to address the inspector-identified deficiency. The WO had already been closed, but the set-screw was tightened by the licensee per the instructions in the completed WO.

In accordance with licensee procedure NPG-SPP-09.3, Plant Modifications and Engineering Change Control, there was a requirement for a post-issuance change (PIC) to the engineering change package (ECP) because the ice basket repairs could not be accomplished with the design provided, but no PIC was requested or issued until the inspectors identified that the lower bracket groves in the lower ice basket supports were not designed wide enough to accommodate the basket end assembly material thickness. WO implementation is addressed by licensee procedure MMDP-1, Maintenance Management System, and requires that plant modifications are performed in accordance with NPG-SPP-09.3. Work per the WO in this case was not performed per the approved ECP because set screws were left loose on the recently installed lower brackets and an insufficient number of wire ropes were installed on damaged basket D7 in Bay 2. In both cases, these deficiencies were identified by the inspectors after the work was signed off as complete and following Quality Control inspection and acceptance.

Analysis: The licensee's failure to adequately develop and implement ice condenser ice basket repairs in accordance with approved engineering and maintenance documents was a performance deficiency. The inspectors reviewed IMC 0612 and determined that the finding was more than minor because, if left uncorrected, it would have the potential to lead to a more significant safety concern; specifically, the lack of structural integrity of the affected ice baskets. The inspectors determined that the failure to adequately repair damaged ice condenser baskets would have a direct effect upon the operability, availability, and reliability of a mitigating system. The finding was associated with the design attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that mitigate plant transients and the reactor accidents. Specifically, without the specified repairs being properly implemented on the damaged ice baskets, there was a potential for damage to adjacent ice baskets obstructing open flow paths or basket ejection, in the event the ice condenser was required to perform its design function. The inspectors determined that the finding was of very low safety significance (Green) because no actual loss of safety function occurred. The cause of the finding had a cross-cutting area of Human Performance in the aspect of effective supervisory/management oversight in the Work Practices component. Specifically, the licensee failed to provide adequate supervisory and management oversight of work activities, including the licensee engineering personnel that prepared and reviewed the ECP, the contractors that performed the repair work and the Quality Control personnel that performed the repair inspection and acceptance. (H.4 (c)).

Enforcement: 10 CFR 50 Appendix B, Criterion III, Design Control, states in part that measures shall be established to assure that regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. In addition, design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Contrary to the above, the licensee failed to adequately develop and implement ice condenser ice basket repairs. Because the finding was of very low safety significance and has been entered into the licensee's corrective action program as PER 623040, this violation was treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy and is identified as NCV 05000390/2012005-02 Failure to Adequately Develop and Implement Ice Condenser Ice Basket Repairs.

.2 Introduction: Engineering justification for the use of an engineering document change (EDC) and an equivalent change (EQV) for hardware modifications to ice baskets that were nonconforming.

Description: The inspectors reviewed outage WO 113393057 which specified the installation of new hardware on a total of six ice baskets that had been damaged, apparently due to ice condenser maintenance. The ice condenser is located within the primary containment and was designed, tested, qualified and fabricated by Westinghouse, the nuclear steam supply system original equipment manufacturer (OEM). The ice condenser contains a total of 1944 vertically supported, perforated 14 gauge sheet metal, ice baskets (12.1 inches in diameter and 48 feet tall) each weighing a maximum of 2200 pounds (ice column + basket). Each of the six damaged baskets (non-conforming components) had suffered plastic deformation (local compressive buckling) of support ligaments in the vicinity of the bottom three feet of the baskets. Instead of replacing the damaged portion of the baskets, as permitted by FSAR Section 6.7.4, licensee engineering had designed hardware to add to the damaged portions of the baskets. EDC E-50607, Revision A, specified the installation of vertical supports mechanically attached on the outside of the damaged area and EQV 60275, Revision A, specified the use of wire rope (steel cable) laced through the damaged area. According to the referenced calculation, WCG-1-1912, Qualification of the Optional Lower Ice Basket Support, the vertical supports were intended for compressive loading and the wire rope was intended for tensile loading. Per FSAR Table 6.7-2, during a deadweight load or deadweight and seismic loads the vertical load on the ice baskets is in compression. When subjected to a design basis accident (DBA) load in combination with a deadweight, or deadweight and earthquake load, the vertical load on all the ice baskets is in tension and the compressed ice basket would tend to elongate.

Review of FSAR Section 6.7.4.3, Design Evaluation, Loading Conditions, part 2., Blowdown Loads, subpart E. Horizontal Ice Basket Forces, states that the tangential and radial forces acting on the ice baskets due to cross flow are assumed to act on the bottom, three feet of ice basket (one-half of the span between the top of the lower support structure and the attachment of the ice baskets to the first lattice frame). The inspectors did not find that the licensee-developed design changes adequately considered these dynamic tangential and radial loads on the damaged ice baskets. Their modifications only addressed either tensile or compressive forces on the ice baskets. Also the addition of the hardware appeared to be more appropriately governed by the requirements of a temporary alteration control form (TACF) per procedure NPG-SPP-09.5, Temporary Alterations, since information contained in the WO indicated the damaged baskets would have to be replaced during the next refueling outage. In addition, the 10CFR50.59 screening processes employed for the addition of hardware did not adequately consider the key elements that would be addressed for a TACF, since hardware was being added to safety-related components. The modifications may not be adequate for the damaged ice baskets to withstand all static and dynamic loads they were originally designed, tested, and qualified to be subjected to. Although there appears to have been some verbal contact between the licensee and the OEM engineering organization regarding the damaged ice baskets, there was no formal OEM review and acceptance of the licensee modifications as an acceptable alternative to ice basket replacement or repair per FSAR Section 6.7.4. Pending additional information from the licensee which can verify that there was adequate engineering justification for the use of an EDC and an EQV for hardware modifications to ice baskets, this item is identified as unresolved item (URI) 050000390/2012005-03, Engineering Justification for Modifications to Non-Conforming Ice Baskets.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed four 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 NPG-SPP-06.9, Testing Programs; NPG-SPP-06.3, Pre-/Post-Maintenance Testing; and NPG-SPP-07.1, On Line Work Management.

- WO 112673625, 1-SI-70-701, Containment isolation valve local leak rate test - component cooling water
- WO 113923819, Component cooling system (CCS) thermal relief check valve open testing with flow
- WO 114243614, Steam generator (SG) 2 main feedwater regulating valve failing to control SG level
- WO 114239806, SG 3 main feedwater regulating valve – replaced pneumatic controller per system engineering request

b. Findings

No findings were identified.

1R20 Refueling and Outage (RFO) Activities

a. Inspection Scope

The licensee continued its U1C11 RFO from the beginning of the reporting period until the unit was returned to 100 percent power on November 5, 2012. The inspectors observed mode changes, portions of the plant heatup, reactor startup and power ascension.

The inspectors monitored licensee controls over the outage activities listed below. In addition, the inspectors reviewed the licensee's corrective action program to ensure that the licensee was identifying equipment alignment problems and that they were properly addressed for resolution.

- 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 verified by reviewing RCS leakage calculations; and containment integrity verified by reviewing the status of containment penetrations and containment isolation valves
- Containment closure activities, including a detailed containment walkdown prior to startup, to verify no evidence of leakage and that debris had not been left which could affect the performance of the containment sump or ice condenser
- Licensee management of fatigue by reviewing schedules, time sheets, and waivers to manage fatigue and associated administrative controls.

b. Findings

No findings 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; NPG-SPP-06.9, Testing Programs; NPG-SPP-06.9.2, Surveillance Test Program; and NPG-SPP-09.1, 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.

In-Service Test:

- WO 112674493, 1-SI-3-923-A, Auxiliary feedwater pump 1A-A comprehensive pump test

RCS Leak Detection

- WO 113238500, 1-SI-90-19, 92-day cot of containment building upper compartment particulate radiation monitor Loop 1-LPR-90-112A

Ice Condenser

- WO 112673482, 1-SI-61-7, 18 month ice condenser intermediate deck doors operational check
- WO 112673484, 1-SI-61-9, 18 month ice condenser floor drains visual examination
- WO 112673481, 1-SI-61-5, 18 month ice condenser lower inlet doors inspection

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Evaluation

a. Inspection Scope

The inspectors evaluated the adequacy of the licensee's methods for testing the alert and notification system in accordance with NRC Inspection Procedure 71114, Attachment 02, Alert and Notification System Evaluation. The applicable planning standard, 10 CFR Part 50.47(b)(5) and its related 10 CFR Part 50, Appendix E, Section IV.D, requirements were used as reference criteria. The criteria contained in NUREG-0654, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, Revision 1, were also used as a reference.

Documents reviewed are listed in the Attachment. Inspectors also observed conduct of a daily siren polling. This inspection activity satisfied one inspection sample for the alert and notification system on a biennial basis.

b. Findings

No findings were identified.

1EP3 Emergency Response Organization Staffing and Augmentation System

a. Inspection Scope

The inspectors reviewed the licensee's Emergency Response Organization (ERO) augmentation staffing requirements and process for notifying the ERO to ensure the readiness of key staff for responding to an event and timely facility activation. The qualification records of key position ERO personnel were reviewed to ensure all ERO qualifications were current. A sample of problems identified from augmentation drills or system tests performed since the last inspection was reviewed to assess the effectiveness of corrective actions.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 03, Emergency Response Organization Staffing and Augmentation System. The applicable planning standard, 10 CFR 50.47(b)(2), and its related 10 CFR 50 Appendix E requirements, were used as reference criteria.

Documents reviewed are listed in the Attachment. This inspection activity satisfied one inspection sample for the ERO staffing and augmentation system on a biennial basis.

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The NSIR headquarters staff performed an in-office review of the latest revisions of various Emergency Plan Implementing Procedures (EPIPs) and the Emergency Plan located under ADAMS accession numbers ML12199A022, ML12296A649, ML12307A285, and ML12313A519, as listed in the Attachment.

The licensee determined that in accordance with 10 CFR 50.54(q), the changes made in the revisions resulted in no reduction in the effectiveness of the Plan, and that the revised Plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The NRC review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, these revisions are subject to future inspection. Documents reviewed are listed in the Attachment. This inspection activity satisfied one inspection sample for the emergency action level and emergency plan changes on an annual basis.

b. Findings

No findings were identified.

1EP5 Maintenance of Emergency Preparedness

a. Inspection Scope

The inspectors reviewed the corrective actions identified through the Emergency Preparedness program to determine the significance of the issues, the completeness and effectiveness of corrective actions, and to determine if issues were recurring. The licensee’s post-event after action reports, self-assessments, and audits were reviewed to assess the licensee’s ability to be self-critical, thus avoiding complacency and degradation of their emergency preparedness program. The licensee’s 10 CFR 50.54(q) change process and selected evaluations of Emergency Preparedness document revisions were reviewed to assess adequacy. The inspectors toured facilities and reviewed equipment and facility maintenance records to assess licensee's adequacy in maintaining them and observed a station Emergency Response Oversight Committee meeting. In addition, the inspectors reviewed licensee procedures and training for the evaluation of changes to the emergency plan.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 05, "Maintenance of Emergency Preparedness." The applicable 10 CFR 50.47(b) planning standards and related 10 CFR 50 Appendix E requirements were used as reference criteria.

Documents reviewed are listed in the Attachment. This inspection activity satisfied one inspection sample for the maintenance of emergency preparedness on a biennial basis.

b. Findings

Introduction: The inspectors identified a self-revealing non-cited violation(NCV) of 10 CFR 50.54(q)(2) for failing to follow the approved emergency plan. Specifically, on August 10, 2012, State of Tennessee officials were not notified within15 minutes of the declaration of an Unusual Event, a risk-significant planning standard requirement of 10 CFR 50.47(b)(5) and 10 CFR 50 Appendix E Section IV.D.3 and Section 5.2.1 of the licensee's Radiological Emergency Plan. The finding was determined to be of very low safety significance (Green).

Description: On August 10, 2012, at 0834, main control room operators received a report of a strong ammonia smell in the turbine building. All personnel were evacuated and access restricted to those wearing appropriate breathing protection. The shift manager who normally initially assumes duties as site emergency director during an event was away from the main control room attending a daily planning meeting and had turned over command and control to the unit supervisor, a qualified senior reactor operator. The unit supervisor evaluated plant conditions using Emergency Plan Implementing Procedure (EPIP)-1,"Emergency Plan Classification Logic", Revision 37, and determined that the Notice of Unusual Event (NOUE) threshold for Emergency Action Level (EAL) 4.4, Toxic Gas, was met. The criterion met was "Normal Operations impeded due to access restriction caused by toxic gas concentrations within a Facility Structure listed in Table 4-2". Table 4-2 included the turbine building. The unit supervisor appropriately declared a NOUE at 0848. Actions were taken to secure the source of the ammonia spill and it was reported stopped at 0851. The unit supervisor performed EPIP-2, Notification of Unusual Event, Revision 30, which had been revised two months prior. Although aware of the requirement to notify state officials within 15 minutes of making any emergency declaration, the unit supervisor was not proficient in performing some steps of the revised EPIP-2 and did not prioritize his activities such that the Tennessee Emergency Management Agency was not notified until 0907, 19 minutes after the declaration. The event did not result in any plant transient. The licensee restored unrestricted access to the turbine building and exited the Unusual Event at 1325.

Analysis: The inspectors determined that failing to notify state officials within 15 minutes of declaring an Unusual Event as required by the Radiological Emergency Plan was a performance deficiency. The finding was greater than minor because it was associated with the Emergency Preparedness cornerstone attribute of ERO performance during an actual event. The finding affected the cornerstone objective in that timely notification is critical to ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The inspectors reviewed this finding using IMC 0609, Appendix B, Emergency Preparedness Significance Determination Process, Attachment 1, "Failure to Implement (Actual Event) Significance Logic." The finding was determined to be of very low safety significance because it was a failure to implement during an Unusual Event. The finding had a cross-cutting aspect in the area of Human Performance, Decision-Making component, because the unit supervisor, in the absence of the shift manager, did not effectively fulfill his responsibility to direct or perform required state communications within the required 15 minute time period as required by the Radiological Emergency Plan. (H.1(a))

Enforcement: 10 CFR 50.54(q)(2) required that a licensee shall follow an emergency plan that meets the requirements of Appendix E and the planning standards of 10 CFR 50.47(b). 10 CFR 47(b)(5) requires that procedures have been established for licensee notification of state response organizations. Additionally, 10 CFR 50, Appendix E, Section IV.D.3, states that licensees shall have the capability of making such notification within 15 minutes after declaring an emergency. The licensee's Radiological Emergency Plan, Section 5.2.1, included that requirement. Contrary to the above, on August 10, 2012, the licensee failed to notify state response organizations within 15 minutes of declaring an Unusual Event. Specifically, the licensee declared an Unusual Event at 0848 and notified state response organizations at 0907, 19 minutes after the declaration. Immediate corrective actions included removing the unit supervisor from watch-standing, subsequently remediated and returned to watchstanding duties. Additional actions included a discretionary operations department clock reset to highlight the event. A Standing Order was issued reinforcing emergency plan notification requirements when assuming shift manager responsibilities. The event was discussed at length with the other TVA nuclear licensees. An apparent cause investigation was performed to determine apparent and contributing causes and EPIP-2 was subsequently revised to improve the reporting process. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance and was entered into their corrective action program as PER 595200 to address recurrence, NCV 05000390/2012005-04, Late State Notification of Unusual Event

1EP6 Drill Evaluation

a. Inspection Scope

Inspectors evaluated the conduct of routine licensee emergency drill on December 11, 2012, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room and Technical Support Center to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Plan Classification Matrix, Revision 37. The inspectors also attended the licensee critique of the drill to compare any inspector observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying deficiencies. The inspectors completed one sample.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee submittals relative to the PIs listed below for the period July 1, 2011, and June 30, 2012. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, were used to confirm the reporting basis for each data element.

Emergency Preparedness Cornerstone

- Drill/Exercise Performance (DEP)
- Emergency Response Organization Drill Participation (ERO)
- Alert and Notification System Reliability (ANS)

For the specified review period, the inspector examined data reported to the NRC, procedural guidance for reporting PI information, and records used by the licensee to identify potential PI occurrences. The inspectors verified the accuracy of the PI for ERO drill and exercise performance through review of a sample of drill and event records. The inspectors reviewed selected training records to verify the accuracy of the PI for ERO drill participation for personnel assigned to key positions in the ERO. The inspectors verified the accuracy of the PI for alert and notification system reliability through review of a sample of the licensee's records of periodic system tests. The inspectors also interviewed the licensee personnel who were responsible for collecting and evaluating the PI data. Documents reviewed are listed in the Attachment. This inspection satisfied three inspection samples for PI verification on an annual basis.

b. Findings

No findings were identified.

4OA2 Identification & Resolution of Problems

.1 Review of Items Entered into the Corrective Action Program (CAP)

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 Semi-Annual Review to Identify Trends

a. Inspection Scope

As required by IP 71152, Identification and Resolution of Problems, the inspectors performed a review of the licensee's corrective action program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on human performance trends, licensee trending efforts, and repetitive equipment and corrective maintenance issues. The inspectors also considered the results of the daily inspector CAP item screening discussed in Section 4OA2.1. The inspectors' review nominally considered the six-month period of July 2012 through December 2012, although some examples expanded beyond those dates when the scope of the trend warranted.

b. Observations

No findings were identified. However, as a result of the below listed annual sample, the licensee has determined that some PERs associated with other NCVs may have a similar weakness. The licensee is currently in the process of creating additional corrective action documents to ensure that all NCVs over the past three years have adequate corrective actions. Also, review of the corrective action performance metrics indicates some amount of backlog build-up.

.2 Annual Sample: Review of the Corrective Actions Associated with NCV 05000390/2010002-01, Failure to Assure That Test Requirements Were Satisfied Following a Design Change.

a. Inspection Scope

Inspectors reviewed PER 215224 which had been credited in NCV 05000390/2010002-01.

b. Findings and Observations

This NCV was issued to the licensee for returning a component to service without meeting the associated post maintenance testing acceptance criteria.

PER 215224 had four corrective actions: 1) Replace the switch which was being tested, 2) Perform failure analysis, 3) Act on results of the failure analysis and 4) Close PER 215224. This PER is shown in the data base as complete. None of these enumerated corrective actions addressed the cause of the original NCV. However, detailed review of immediate actions indicated that some level of coaching of the procedure performer was done. Additional follow-up with licensee personnel indicated that this event had later been used during pre-job briefings relating to procedure usage. Therefore, the corrective actions in total were found to be weak, but adequate.

4OA3 Event Follow-up

.1 Unit 1 NOUE – August 10, 2012

a. Inspection Scope

The inspectors responded to a Unit 1 due to a release of ammonia in the Unit 1 turbine building from an overflowed ammonia mixing tank. The tank overflowed due to leakage past the seat of a not fully closed valve. The licensee conservatively declared a Notification of Unusual Event (NOUE) due to inaccessibility of some areas of the turbine building. The inspectors reviewed the initial licensee event notification to verify that it met regulatory requirements.

b. Findings

No findings were identified.

.2 Isolation valve FCV-15-44, steam generator blowdown valve to the cooling tower blowdown line failed to close on process radiation alarm signal

a. Inspection Scope

Inspection reviewed activities associated with failure of FCV-15-44.

b. Findings

Introduction: A Green, self-revealing NCV of TS 5.7.1, Procedures, was identified for failing to adhere to OPDP-1, Conduct of Operations Section 5.1, Procedure Adherence.

Description. On October 26, 2012, a spurious high radiation alarm was received in the main control room (MCR) for the steam generator blow down (SGBD) radiation monitor. Counts were verified less than the set point; however, it was noted that SGBD flow path safety feature had not actuated as expected. A request was submitted for troubleshooting and repairs by the instrument maintenance group (MIG). In the course of troubleshooting on November 1, 2012, why automatic actions did not occur on October 26, 2012, an MIG technician unexpectedly discovered an electrical jumper already installed in the position where he was about to place his troubleshooting jumper.

The MIG technician stopped work and notified MIG supervision. The jumper log revealed that this electrical jumper was installed on Sep 10, 2012, at 0530.

Investigation by the licensee revealed that a reactor operator had failed to follow System Operating Instruction (SOI)-90.01, Rev. 29, Liquid Process Radiation Monitors, step 5.5 [10], which is a conditional step to remove a jumper, if it was installed in a previous step. The jumper had been installed at the beginning of a refueling outage some weeks earlier. The reactor operator failed to use the jumper log to determine if the jumper was installed. Rather, the reactor operator called the maintenance instrument shop and inquired as to the jumper status. Based on the results of the call, he assumed the jumper had been removed and placed the SGBD system in service on October 24, 2012. On October 26, 2012, operators noted a high radiation alarm on the system and that the SGBD valve to the cooling tower blowdown line failed to shut. SR 631272 was written to troubleshoot and repair. On November 01, 2012, maintenance personnel found the jumper installed which prevented trip on high radiation of 1-FCV-15-44, SGBD valve to the cooling tower blowdown line.

Analysis. The inspectors determined that the failure to properly implement procedure SOI-90.01, Liquid Process Radiation Monitors, Rev. 29, was more than minor because, if left uncorrected, it could lead to a more significant safety issue, a radioactive release, and was also associated with the Mitigating Systems Cornerstone attribute of equipment performance (reliability) and adversely affected the cornerstone objective. The inspectors evaluated the risk significance of this finding using IMC 0609, Significance Determination Process Phase 1. The finding screened to very low safety significance (Green) because actual high contamination levels did not occur within the steam generators during the period that the jumper was installed. The finding directly involved the cross-cutting area of Human Performance under the procedural compliance aspect of the Work Practices component; in that the procedural requirements of SOI-90.01 were not met. (H.4(b))

Enforcement. TS 5.7.1.1.a requires that written procedures shall be implemented and maintained covering the activities in the applicable procedures recommended by Regulatory Guide (RG) 1.33, Revision 2, Appendix A, of which part 7, Procedures for Control of Radioactivity (for limiting materials released to environment and limiting personnel exposure), should be covered by written procedures. Contrary to this requirement, the licensee did not properly implement procedural requirements for placing the SGBD system into service. Because this violation was of very low safety significance and was entered into the corrective action program (PER 637279), this violation is being treated as a non-cited violation (NCV) consistent with Section 2.3.2. of the Enforcement Policy: NCV 05000390/2012005-05, Failure to Follow Procedure Resulted in Failing to Remove Jumpers Inhibiting Proper Operation of the Steam Generator Blowdown System.

4OA5 Other Activities

.1 (Closed) Temporary Instruction 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

The inspectors accompanied the licensee on their seismic walkdowns of the intake pumping station on July 24, 2012 and the control and auxiliary buildings on August 7, 2012, and verified that the licensee confirmed that the following seismic features associated with the ERCW pumps and motors and ERCW strainers were free of potential adverse seismic conditions.

- Anchorage was free of bent, broken, missing, or loose hardware.
- Anchorage was free of corrosion that is more than mild surface oxidation.
- Anchorage was free of visible cracks in the concrete near the anchors.
- Anchorage configuration was consistent with plant documentation.
- Structures, systems, or components (SSCs) will not be damaged from impact by nearby equipment or structures.
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment.
- Attached lines have adequate flexibility to avoid damage.
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area.
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area.
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

On August 14, 2012, the inspectors independently performed their walkdown and verified that the following 1B residual heat removal system, including the pump, motor, heat exchanger and associated valves, all located in the auxiliary building were free of potential seismic conditions.

- Anchorage was free of bent, broken, missing, or loose hardware.
- Anchorage was free of corrosion that is more than mild surface oxidation.
- Anchorage was free of visible cracks in the concrete near the anchors.
- Anchorage configuration was consistent with plant documentation.
- SSCs will not be damaged from impact by nearby equipment or structures.
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment.
- Attached lines have adequate flexibility to avoid damage.
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area.

- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area.
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

Observations made during the walkdown that could not be determined to be acceptable were entered into the licensee's corrective action program for evaluation.

Additionally, inspectors verified that items that could allow the spent fuel pool to drain down rapidly were added to the Seismic Walkdown Equipment List (SWEL), and these items were walked down by the licensee.

b. Findings

No findings were identified.

.2 (Closed) Unresolved Item (URI) 05000390/2010007-02, Installed Insulating Fluid in Interior Transformers Potentially Deviates from License/ Design Criterion in SSER 18 and Position D.1.g of Appendix A to BTP (APCSB) 9.5-1.

a. Inspection Scope

This URI was opened for resolution of issues pertaining to Watts Bar (WBN) Unit 1 licensing basis for the installation of dielectric insulating liquid in indoor power transformers. During an NRC Triennial Fire Protection Inspection (TFPI), as documented in NRC Inspection Report 05000390/2010007, the inspectors identified that the licensee had replaced an Askarel-type transformer dielectric insulating liquid within indoor power transformers with a "high fire point," combustible silicone-type insulating fluid [Dow Corning 561® (DC 561)]. During the WBN, Unit 1, 2010 TFPI, the NRC questioned if the installed "high fire point," combustible silicone-type insulating fluid was a non-compliance and if the transformer insulating material was consistent with the NRC approved licensing basis criteria described in WBN SSER 18, section 5.10.2, "Askarel-Insulated Transformers" and NRC Position D.1.g of Appendix A to (BTP) APCSB 9.5-I, "Guidelines for Fire Protection for Nuclear Power Plants," dated August 23, 1976. The issue remained opened pending further NRC review of information related to the plant fire protection licensing/design basis.

Subsequent to the 2010 TFPI at WBN, TVA provided to the Fire Protection Branch in the Office of Nuclear Reactor Regulation (NRR) additional information regarding TVA's use of "high fire point" silicone insulating fluid in indoor transformers in lieu of the non-combustible Askarel-type liquid described in SSER 18, section 5.10.2, and Appendix A to (BTP) APCSB 9.5-1. Included in the provided information were responses to NRR staff's requests for additional information (RAIs) contained in TVA letters dated August 5, 2011, and September 30, 2011. TVA also provided to the Region II inspection staff additional information regarding specific sprinkler system flow densities in plant areas containing "high fire point" silicone insulating fluid filled indoor transformers; TVA's conformance to DC 561 Underwriters Laboratories (UL) Classification Marking; and prescribed vendor recommendations for use of DC 561silicone insulating fluid in indoor transformers identified in Section 4.3.2 of the DC 561 technical manual.

An in-office review was performed by the Region II inspectors of Watts Bar's responses to the NRR; Region II RAIs regarding silicone insulating fluid in indoor transformers at WBN; and the WBN fire protection program licensing and design basis documents. The NRC determined that Watts Bar did not violate any of their fire protection licensing or design basis documents relative to installation of "high fire point," silicone-type insulating liquid in indoor power transformers.

b. Findings

No findings were identified.

.3 Temporary Instruction 2515/187 – Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

Inspector(s) verified that licensee's walkdown packages, related to walkdowns in the intake pumping station and the auxiliary building flood mode spool pieces, contained the elements as specified in NEI 12-07, Walkdown Guidance document:

The week of July 23, 2012, the inspectors accompanied the licensee on their walkdown of those activities related to accomplishing the plant's flood mode strategy and verified that the licensee confirmed the following flood protection features:

- Visual inspection of the flood protection feature was performed if the flood protection feature was relevant. External visual inspection for indications of degradation that would prevent its credited function from being performed was performed.
- Reasonable simulation, if applicable to the site
- Critical SSC dimensions were measured
- Available physical margin, where applicable, was determined.
- Flood protection features which included spool piece simulated installations were observed to verify licensee capability to meet installation time requirements.

The inspectors independently performed their walkdown and verified that the following flood protection features were in place:

- Installed HESCO barriers were reviewed for material condition
- Review of installation of HESCO barriers including demonstrations to fill gaps
- Flood protection features in the intake pumping station were reviewed, including seals, sump pumps, check valves and equipment failure histories.
- Available physical margin, where applicable, was determined.

The inspectors verified that noncompliances with current licensing requirements, and issues identified in accordance with the 10 CFR 50.54(f) letter, Item 2.g of Enclosure 4, were entered into the licensee's corrective action program. In addition, issues identified in response to Item 2.g that could challenge risk significant equipment and the licensee's ability to mitigate the consequences will be subject to additional NRC evaluation.

b. Findings

No findings were identified

4OA6 Meetings, including Exit

On October 19, 2012, the lead EP inspector presented the inspection results to Mr. D. Grissette, and other members of the staff. The inspector confirmed that proprietary information was not provided or reviewed during the inspection.

On January 11, 2012**,** the resident inspectors presented the quarterly inspection results to Mr. Don Grissette, Site Vice President, and other members of the licensee staff. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which met the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

- Technical Requirements 3.4.2, Pressurizer Temperature Limits, required that the pressurizer heatup rate be limited to ≤100° F in any 1-hour period. The TR action statement A.1 required that the limit be restored within 30 minutes. Contrary to the above, at or around 0834, on October 10, 2012, while Unit 1 was in Mode 5, the licensee determined that the pressurizer vapor space heatup rate limit of ≤100° F in any 1-hour period had been exceeded. The heatup rate was 103° F per hour. The heatup rate was returned to within limits in less than the limiting condition of operation (LCO) action time of 30 minutes. The finding was screened in accordance with IMC 0609 Appendix G, Shutdown Operations SDP and was characterized to be of very low safety significance (Green) because the pressurizer water temperature did not exceed the TR heatup rate, only the vapor temperature by a small (3° F per hour) amount and the engineering staff review using OEM documentation concluded that there were no adverse consequences.

- 10 CFR 50 Appendix B, Criterion III, Design Control, states in part that measures shall be established to assure that regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the licensee failed to control both ASME III code and non-ASME code materials during relocation of WBN-1-PDT-030-0042-G, WBN-1-PDT-030-0045-D, and WBN-1-PDT-030-0043-F installed by Design Change Notice 58382 stages 8, 10, and 11. Portions of the installed material did not met ASME Section III

Class 2, TVA Class B design requirements resulting in non-ASME code material being used in the fabrication of ASME code components. The finding was screened in accordance with IMC 0609 Appendix G, Shutdown Operations SDP and was characterized to be of very low safety significance (Green) because the non-conforming condition documentation concluded that there were no adverse functional consequences.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
R. Bankes, (Interim) Chemistry/Environmental Manager
T. Carter, (interim) Site Engineering Director
T. Cleary, Interim Site Vice President
T. Detchemende, Emergency Preparedness Manager
R. Dittmer, Operations Superintendent
W. Francis, (Interim) Maintenance Manager
D. Gronek, Plant Manager
D. Guinn, Licensing Manager
E. Higgins, Civil Design Manager
W. Hooks, Radiation Protection Manager
D. Hughes, Training Supervisor
B. Hunt, Operations Support Superintendent
D. Jacques, Security Manager
R. Kirkpatrick, Design Engineering Manager
W. Prevatt, Operations Manager
A. Scales, Work Control Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

050000390/2012005-01	URI	Engineering Justification for Design of Control Building Watertight Hatches (Section 1R06)
050000390/2012005-03	URI	Engineering Justification for Modifications to Non-Conforming Ice Baskets (Section 1R18.2)

Opened and Closed

05000390/2012005-02	NCV	Failure to Adequately Develop and Implement Ice Condenser Ice Basket Repairs (Section 1R18.1)
05000390/2012005-04	NCV	Late State Notification of Unusual Event (Section 1EP5)
05000390/2012002-05	NCV	Failure to Follow Procedure Resulted in Failing to Remove Jumpers Inhibiting Proper Operation of the Steam Generator Blowdown System (Section 4OA3)

Closed

05000390/2010007-02	URI	Installed Insulating Fluid in Interior Transformers Potentially Deviates from License/Design Criterion in SSER 18 and Position D.1.g of Appendix A to BTP(APSCB) 9.5-1 (Section 4OA5)
2515/188	TI	Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns (Section 4OA5.2)

Discussed

2515/187	TI	Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Section 1R01: External Flood Protection Inspection
WO 113273211, Procedure 1-PI-OPS-1-FP, Freeze Protection, Rev. 0040

Section 1R04: Equipment Alignment
SOI-72.01, Containment Spray System Power Checklist 72.01-1P
SOI-72.01, Containment Spray System Valve Checklist 72.01-1V
SOI-74-01, Residual Heat Removal System Power Checklist 74.01-1P
SOI-74-01, Residual Heat Removal System Power Checklist 74.01-2P
SOI-74-01, Residual Heat Removal System Power Checklist 74.01-3P
SOI-74-01, Residual Heat Removal System Valve Checklist 74.01-1V
SOI-74-01, Residual Heat Removal System Valve Checklist 74.01-2V
SOI-74-01, Residual Heat Removal System Valve Checklist 74.01-3V
SOI-3.02, Auxiliary Feedwater System Power Checklist 3.02-1P
SOI-3.02, Auxiliary Feedwater System Valve Checklist 3.02-1V

Section 1R06: Flood Protection Measures
Updated Final Safety Analysis Report (UFSAR) Section 3.8.4.1.1
NPG Calculation MDN-000-999-2008-0145, WBN Probabilistic Risk Assessment – Internal Flooding Analysis Notebook, Rev. 1
NPG Calculation WBNAPS2-165, Turbine Building Flooding Due to a Break in the Condenser Circulating Water System, Rev. 4
NPG Calculation WCGE023, Review of Flood Protection Requirements for Watertight Doors and Hatches, Rev. 3
Design Criteria Document WB-DC-40-60, Special Hatches and Manways, Rev. 6
Design Criteria Document WB-DC-20-21, Miscellaneous Steel Components for Category I Structures, Rev. 13
WBN Maintenance Instruction MI-270-07, Visual Examination of Control and Auxiliary Building Doors and Hatchways, Rev. 7

Ebasco Services, Inc. Calculation WCG-1-1591, "Seismic Evaluation of Watertight Equipment Hatches in Control Building at El. 708.0, Rev. 0
PER 99-008763-00
Service Request (SR) 427917
Work Order (WO) 112678945

Section 1EP2: Alert and Notification System Evaluation
Procedures and Reports
EPDP-10, Facilitation of the Alert and Notification System and Pager Tests, Rev. 5
EPDP-14, Evaluation of Changes to Alert and Notification Systems (ANS), Rev. 0
EPDP-17 Nuclear Power Group (NPG) Emergency Plan Effectiveness Review [10 CFR 50.54(q)], Rev. 0001
EPFS-9, Inspection, Service, and Maintenance of the Prompt Notification System (PNS) at Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants, Rev. 0007
TVA Nuclear Power Radiological Emergency Plan, Generic, Revision 97
TVA Nuclear Power Radiological Emergency Plan, Appendix C, Watts Bar Nuclear Plant, Rev. 97

Records and Data
Monthly and Bi-weekly Activation Results, January 1, 2011 – December 31, 2011
Monthly and Bi-weekly Activation Results, January 1, 2012 – September 30, 2012
PNS Trouble Reports January 1, 2011 – December 31, 2011
PNS Trouble Reports January 1, 20112 – September 30, 2012
EPDP 17 Screening Evaluation and Effectiveness Form Packet CECC 2012-051
EPDP 17 Screening Evaluation and Effectiveness Form Packet CECC 2012-052
TVA Nuclear Power Group Focused Self-Assessment Report, Assessment No. WBN-EP-F-12-001, NRC Pre-Inspection

Corrective Action Documents
PER 594343; 3 Failed sirens
PER 577441, Failure of seven WBNP ANS sirens

Section 1EP3: Emergency Response Organization Staffing and Augmentation System
Procedures
EPDP-2, Emergency Duty Officer, Emergency Preparedness Staff and Operations Duty Specialist Notification Procedures, Rev. 3
EPDP-10, Facilitation of the Alert and Notification System and Notification Tests, Rev. 5

Records and Data
EPDP-10, Facilitation of the Alert and Notification System and Notification Tests, Rev. 2,
Attachment 1, Pager Test Performance Documentation, January 2011 – May 2012
EPDP-10, Facilitation of the Alert and Notification System and Notification Tests, Rev. 4,
Attachment 1, Notification Test/Operability Check Performance Documentation,
June – September 2012
Selected TVA Automated Training Information System Employee Transcript Records for Emergency Preparedness Duty List Members from November 2011 to August 2012
Watts Bar Station Emergency Response Organization Emergency Preparedness Duty List, dated 10/9/12

Snapshot Self-Assessment Report, Assessment Report No. WBN-EP-S-11-006, ERO Pager System
Snapshot Self-Assessment Report, Assessment Report No. WBN-EP-S-12-002, ERO Participation Assessment
TVA Nuclear Power Group Focused Self-Assessment Report, Assessment No. WBN-EP-F-12-001, NRC Pre-Inspection
Corrective Action Documents
PER 298366, Trend noting WBN REP Pager test regarding pager batteries
PER 438540, Evaluate three individuals living over an hour from site for being on ERO team

Section 1EP4: Emergency Action Level and Emergency Plan Changes
Change Packages
CECC EPIP-8, "Dose Assessment Staff Activities During Nuclear Plant Radiological Emergencies," Revision 37
Tennessee Valley Authority, Radiological Emergency Plan, Revision 97 and 98
EPIP-1, "Emergency Plan Classification Logic," Revision 37

Section 1EP5: Maintenance of Emergency Preparedness
Procedures
EPDP-1, Procedures, Maps, and Drawings, Rev. 5
EPDP-6, Post Emergency Documentation, Rev. 1
EPDP-7, Review of Agreement Letters and Contracts, Rev. 3
EPDP-8, Emergency Preparedness Quality Assurance, Rev. 2
EPDP-16, Designated Emergency Response Equipment (DERE), Rev. 0
EPDP-17, NPG Emergency Plan Effectiveness Review (10CFR 50.54(q)), Rev. 1
EPIP-1, Emergency Plan Classification Logic, Rev 37
EPIP-2, Notification of Unusual Event, Rev. 30 and 31

Records and Data
QA-WB-12-002, 12-009 and 12-013, Watts Bar Nuclear Plant – Quality Assurance – Oversight Reports for the period of October 1, 2011 - December 31, 2011, January 1, 2012 – March 31, 2012 and April 1, 2012 – June 30, 2012
SSA1203, TVA Quality Assurance - Nuclear Power Group (NPG) – Radiological Emergency Preparedness - Audit Report, May 17, 2012
Watts Bar Event Report, "10/17/2011 NOUE Event Report," dated 10/24/2011
Completed procedure, EPIP-1, "Emergency Plan Classification Logic," Rev. 35
Completed procedure, EPIP-2, "Notification of Unusual Event," Rev. 29
QA-WB-11-004, QA Assessment of January 2011 drill
QA-WB-11-012, QA Mid-Cycle Assessment
WBN-EP-F-11-001, Evaluate readiness of EP for the start-up and operation of Unit 2
WBN-EP-S-11-001, Pager Test Results
WBN-EP-S-11-004, Monthly Inventories
WBN-EP-S-11-007, EP Program pre-NRC Inspection Assessment
WBN-EP-S-12-004, DEP Opportunity Assessment
WBN-EP-F-12-001, EP Program pre-NRC Inspection Assessment
WBN-EP-S-12-002, ERO participation Assessment

0.54(q) screenings/reviews of EPIP-1 Rev. 35 and Rev. 36, EPIP-2 Rev. 30, EPIP-3 Rev. 33, EPIP-5 Rev. 39, EPIP-6 Rev.41, REP Appendix “C” Rev. 94, and REP Generic Section and REP Appendix “C” Rev. 97

Corrective Action Documents
314281, Site expectations regarding post drill briefings are not adhered to
360041, Dose Equivalent Iodine dose conversion factors revision
380964, SED did not classify the SAE within 15 minutes during drill
388939, SED failed to set clear priorities during briefs
392559, Unannounced pager test, performed on 6/19/2011
430163, Web-EOC to solve issue of lack of priorities from SED
430190, E-Plan, pages C-181 and 182 do not agree on number of personnel for all positions
448510, NOUE declaration on 10/17/2011, Investigate operational guidance
541683, Open work orders impacting EP are not being consistently coded
543874, January 2012 ANS PI inaccurate
595200, Late NOUE Communications to the State of Tennessee
626219, NOUE after-action report has admin error
626242, NOUE Apparent Cause Evaluation has incorrect event date
626246, Incorrect CFR reference in EPIP-1 Rev. 36 revision review
626249, 1Q12 PI change not in NRC public website

Section 4OA1: Performance Indicator Verification
Procedures
EPDP-11, Emergency Preparedness Performance Indicators, Rev. 4
EPIP-1, Emergency Plan Classification Logic, Rev. 37
EPIP-2, Notification of Unusual Event, Rev. 29 and Rev. 30
NPG-SPP-02.2, Performance Indicator Program, Rev. 3

Records and Data
NOUE Event Report/Critique October 2011
NOUE Event Report/Critique August 2012
Documentation of Performance Indicator data - July 1, 2011 through June 30, 2012 for DEP, ANS, and ERO
Emergency Preparedness PI Data 2010 Quarters 3 and 4
Emergency Preparedness PI Data 2011 Quarter 1 – Quarter 4
Emergency Preparedness PI Data 2012 Quarters 1 and 2
ANS Reliability Form - NRC ANS PI Data January 1, 2011 – December 31, 2011
ANS Reliability Form - NRC ANS PI Data January 1, 2012 – September 30, 2012
Drill and Exercise Performance records from 7/1/2011 through 6/30/2012
TVA Automated Training Information System, Training records for 10 ERO members
TVA Quality Assurance NPG – Watts Bar Nuclear Plant – Emergency Preparedness Audit Report – SSA1203, May 17, 2012 Third Quarter FY 2011 - Third Quarter FY 2012 Emergency Preparedness (EP) Second Tier Performance Indicator (PI) Reports
TVA Nuclear Power Group Focused Self-Assessment Report, Assessment No. WBN-EP-F-12-001, NRC Pre-Inspection

<u>Corrective Action Documents</u>
PER 479063, DEP classification failure
PER 543874; ANS PI error January 2012

<u>Section 4OA5: Other Activities</u>
NRC Letter to TVA, dated June 11, 2010, “Request for Additional Information Regarding Fire Protection Program” (ADAMS Accession No. ML101550488)
NRC Letter to TVA, dated July 21, 2011, Watts Bar Nuclear Plant, Unit 2 - “Request for Additional Information Regarding Final Safety Analysis Report Amendment Related to Section 9.5.1 'Fire Protection System' Group 6” (ADAMS Accession No. ML111823580)
NRC Letter to TVA, dated September 14, 2011, Watts Bar Nuclear Plant, Unit 2 - “Request for Additional Information Regarding Final Safety Analysis Report Amendment Related to Section 9.5.1 'Fire Protection System' Group 7” (ADAMS Accession No. ML112490474)
TVA Letter to NRC, dated August 5, 2011, Watts Bar Nuclear Plant (WBN) Unit 2 – “TVA Response to Request for additional Information (RAI) Group 6 Regarding Fire Protection Report” (ADAMS Accession No. ML11224A052)
TVA Evaluation, “TVA NUREG 1805 Evaluation of WBN Transformer Silicone Oil,” dated March 9, 2012
PER 265331, Maintenance Program for the 6.9kV / 480V Oil Filled Transformers, dated May 19, 2011
American National Standard Institute, ANSI C84.1, “American National Standard for Electric Power Systems and Equipment-Voltage Ratings (60 Hertz),” 2006 Edition.
Drawing 8278D75, Westinghouse Electric Corporation, Power Center Transformer, Revision 3
Product Technical Information, AP-P04-03L-01E and PI-LPRD00, Qualitrol® Large Pressure Relief Devices, Revision 26613
Vendor Information, Dow Corning 561® Silicone Transformer Liquid Training Manual, dated 2006
Vendor Technical Manual, Dow Corning 561® Silicone Transformer Liquid, dated 2006

LIST OF ACRONYMS

AFW	auxiliary feedwater
CLD	current licensing basis
CS	containment spray
CFR	*Code of Federal Regulations*
DBA	design basis accident
EDC	engineering document change
El.	elevation
ERCW	essential raw cooling water
ERO	Emergency Response Organization
EQV	equivalent change
FE	functional evaluation
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
IP	inspection procedure
MOV	motor-operated valve
NCV	non-cited violation
NEI	Nuclear Energy Institute
NPG-SPP	nuclear power group standard programs and processes
NRC	Nuclear Regulatory Commission
NUREG	(NRC) technical report designation
OEM	original equipment manufacturer
OOS	out of service
PER	problem evaluation report
PDO	prompt determination of operability
PI	performance indicator
RCS	reactor coolant system
RFO	refueling outage
RHR	residual heat removal
RTP	rated thermal power
RTS	return to service
SDP	Significance Determination Process
SG	steam generator
SI	safety injection
SR	service request
SSCs	structures, systems, or components
TACF	temporary alteration control form
TDAFW	turbine-driven auxiliary feedwater
TS	technical specifications
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
WO	work order