UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-1257

February 13, 2017

Mr. Joseph W. Shea, Vice President
Nuclear Regulatory Affairs and
Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT – NUCLEAR REGULATORY COMMISSION INTEGRATED INSPECTION REPORT 05000390/2017004 AND 05000391/2017004

Dear Mr. Shea:

On December 31, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Watts Bar Nuclear Plant, Unit 1 and Unit 2. On January 19, 2018, the NRC inspectors discussed the results of this inspection with Mr. Tom Marshall and other members of your staff. The results of this inspection are documented in the enclosed inspection report.

NRC inspectors documented two findings of very low safety significance (Green) in this report which involved violations of NRC requirements. Further, inspectors documented a licensee-identified violation which was determined to be of very low safety significance in this report. The NRC is treating these violations as noncited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 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.

This letter, its enclosure, and your response (if any) will be available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and in the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Anthony D. Masters, Chief
Reactor Projects Branch 5
Division of Reactor Projects

Docket Nos.:50-390, 50-391
License No.: NPF-90, 96

Enclosure:
IR 05000390/2017004, 05000391/2017004
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-390, 50-391
License Nos.:	NPF-90, NPF-96
Report No.:	05000390/2017004, 05000391/2017004
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Units 1 and 2
Location:	Spring City, TN 37381
Dates:	October 1 through December 31, 2017
Inspectors:	J. Nadel, Senior Resident Inspector J. Hamman, Resident Inspector W. Satterfield, Resident Inspector P. Cooper, Reactor Inspector S. Monarque, Project Engineer M. Riches, Project Engineer M. Kennard, Operations Engineer D. Lanyi, Senior Operations Engineer
Approved by:	Anthony Masters, Chief Reactor Projects Branch 5 Division of Reactor Projects

Enclosure

SUMMARY

IR 05000390/2017004; 05000391/2017004; October 1, 2017 – December 31, 2017; Watts Bar Nuclear Plant; Operability Determinations, Plant Modifications, and Refueling and Outage Activities.

The report covered a three-month period of inspection by the resident inspectors and five regional inspectors.

Two Green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (i.e., Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," (SDP) dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within Cross Cutting Areas," dated December 04, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated November 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 6. Documents reviewed by the inspectors not identified in the Report Details are listed in the Attachment.

Cornerstone: Barrier Integrity

- Green. A self-revealed NCV of 10 *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for inadequacies associated with TVA procedure NPG-SPP-09.5, Temporary Modifications Temporary Configuration Changes, Revision 11. Specifically, a procedural exception allowed a temporary configuration change to be installed in the spent fuel pool without a screening in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments." The change subsequently caused an inadvertent draining of the level of the spent fuel pool to the point the control room received the low level alarm.

  The performance deficiency was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of design control, and it adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding affected the common spent fuel pool and the Barrier Integrity Cornerstone while Unit 1 was at power and Unit 2 was in mode 4. For Unit 1, the inspectors determined that this finding was of very low safety significance (Green) because the finding did not result in a loss of spent fuel pool inventory below the minimum analyzed level in the site-specific licensing basis. For Unit 2, the inspectors determined that this finding was of very low safety significance (Green) because it did not involve an actual reduction in function of hydrogen control for PWR ice condenser containments. The finding had a cross-cutting aspect in the Work Management attribute of the Human Performance area as defined in NRC IMC 0310, because the licensee did not implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. (H.5) (Section 1R18)

- Green. An NRC-identified NCV of very low safety significance associated with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to promptly identify and correct a condition adverse to quality. Specifically, NRC inspectors identified a boric acid leak on the Unit 2 loop 1 hot leg sample valve, 2-SMV-68-548, that had been missed by licensee personnel performing boric acid corrosion control program walkdowns during Unit 2 refueling outage 1 (RFO1).

The performance deficiency was determined to be more than minor because, if left uncorrected, it had the potential to lead to a more significant safety concern. The inspectors performed the significance determination using NRC IMC 0609. The finding affected the Barrier Integrity Cornerstone while Unit 2 was shut down, so IMC 0609 Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings,” dated May 9, 2014, was used to determine that this finding was of very low safety significance (Green) because it did not degrade the ability to isolate a drain down or leakage path. The finding had a cross-cutting aspect in the Work Management attribute of the Human Performance area as defined in NRC IMC 0310, because the licensee did not implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. (H.5) (Section 1R20)

One violation of very low safety or security significance that was identified by the licensee has been reviewed by the NRC. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. This violation and the corrective action tracking number is listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 started the reporting period at 100 percent rated thermal power (RTP) and remained there through the end of the reporting period.

Unit 2 started the reporting period at 100 percent RTP and remained there until October 27, 2017, when it began shutting down for a planned refueling outage. The unit remained shut down until it was restarted after the outage on December 6, 2017. On December 11, 2017, the unit had reached 97 percent RTP, it was manually shut down that same day after multiple control rods fell into the core. The unit remained shut down until it was restarted on December 15, 2017, and it reached 100 percent RTP on December 18, 2017. It remained at 100 percent RTP through the end of the reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

Seasonal Readiness Reviews for Cold Weather

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 0-PI-OPS-1-FP, Freeze Protection, including associated checklist 1, Freeze Protection. Inspectors walked down the intake pumping station, Unit 1 and Unit 2 main steam valve vault rooms, Unit 1 and Unit 2 refueling water storage tanks, and external portions of main feed and main steam piping 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 condition reports (CRs) and work orders (WOs) were assessed for effectiveness and timeliness. This activity constituted one Adverse Weather inspection sample, as defined in Inspection Procedure (IP) 71111.01.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

Partial System Walkdowns

a. Inspection Scope

The inspectors conducted the 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 also included that redundant trains were returned to service properly. The inspectors reviewed the functional system descriptions, the 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. This activity constituted five inspection samples, as defined in IP 71111.04.

- 1A and 1B trains of motor-driven auxiliary feedwater (MDAFW) system
- Unit 1 train of turbine-driven auxiliary feedwater (TDAFW)
- 1A train of containment spray
- 1B train of containment spray
- 2A train of residual heat removal system (RHR) during yellow shutdown risk for Unit 2

b. Findings

No findings were identified.

.2 Complete Walkdown

The inspectors performed a complete system walkdown of the Unit 2 residual heat removal system and support systems to verify proper equipment alignment, to identify any discrepancies that could impact the function of the system and increase risk, and to verify that the licensee properly identified and resolved equipment alignment problems that could cause events or impact the functional capability of the system.

The inspectors reviewed the UFSAR, system procedures, system drawings, and system design documents to determine the correct lineup and then examined system components and their configuration to identify any discrepancies between the existing system equipment lineup and the correct lineup. During the walkdown, the inspectors reviewed the following:

- Valves were correctly positioned and did not exhibit leakage that would impact the functions of any given valve
- Electrical power was available as required
- Major system components were correctly labeled, lubricated, cooled, ventilated, etc.
- Hangers and supports were correctly installed and functional
- Essential support systems were operational
- Ancillary equipment or debris did not interfere with system performance
- Valves were locked as required by the locked valve program
- Visible cabling appeared to be in good material condition

In addition, the inspectors reviewed outstanding maintenance work requests and design issues on the system to determine whether any condition described in those work requests could adversely impact current system operability. This activity constituted one inspection sample, as defined in IP 71111.04.

1R05 Fire Protection (71111.05AQ)

Fire Protection Tours

a. Inspection Scope

The inspectors conducted tours of the 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; and 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 four inspection samples, as defined in IP 71111.05AQ.

- Unit 2 control rod drive equipment room and pressurizer heater transformer room
- Emergency diesel generator building
- Unit 2 upper containment
- Unit 2 lower containment

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

.1 Internal Flooding

The inspectors reviewed related flood analysis documents and walked down the area listed below containing risk-significant structures, systems, and components susceptible to flooding. The inspectors verified that plant design features and plant procedures for flood mitigation were consistent with UFSAR design requirements and the internal flooding analysis assumptions. The inspectors assessed the condition of flood mitigation features such as drains, barriers, curbs, and door seals. In addition, the inspectors verified the licensee was identifying and properly addressing internal flooding issues in the corrective action program. This inspection constituted one inspection sample, as defined in IP 71111.06.

- Auxiliary building levels 676, 692, and 713

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

Non-Destructive Examination Activities and Welding Activities

From November 6, 2017, through November 15, 2017, the inspectors conducted an onsite review of the implementation of the licensee's inservice inspection (ISI) program for Unit 2. The ISI program is designed to monitor degradation of pressure retaining components in vital system boundaries. The scope of this program includes components within the reactor coolant system boundary, risk-significant piping boundaries, and containment system boundaries.

The inspectors reviewed the following non-destructive examination (NDE) activities. These activities were mandated by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code of Record: 2007 Edition with 2008 Addenda). The inspectors evaluated the NDE activities for compliance with the requirements in Section XI and Section V of the ASME Code. The inspectors also evaluated if any identified indications or defects were dispositioned in accordance with either the ASME Code or an NRC-approved alternative requirement. Additionally, the inspectors reviewed the qualifications of the NDE technicians performing the examinations to determine if they were in compliance with ASME Code requirements.

- Ultrasonic Examination (UT), SG-1-C-IR, Nozzle inner radius, ASME Class 2 (reviewed)
- UT, SG-1-H-IR, Nozzle Inner Radius, AMSE Class 2 (reviewed)
- Visual (VT-2) Examination, 58 bottom mounted instrument penetrations and the RPV lower head, performed under ASME Code Case N-722-1 (reviewed)

The inspectors reviewed the following welding activities. The inspectors evaluated these activities for compliance with site procedures and the requirements in Section IX and Section XI of the ASME Code. Specifically, the inspectors reviewed the WOs, repair or replacement plans, weld data sheets, welding procedures, procedure qualification records, welder performance qualification records, and NDE reports.

- 2-CKV-063-0557, Replace 2-inch Valve, ASME Code Class I
- 2-CKV-063-0553, Replace 2-inch Valve, ASME Code Class I

The inspectors reviewed the listing of non-destructive surface and volumetric examinations performed during the previous refueling outage. The inspectors verified that the licensee did not identify any relevant indications that were analytically evaluated and accepted for continued service.

PWR Vessel Upper Head Penetration Inspection Activities

The inspectors performed the following activities to verify that the requirements of the ASME Code and applicable licensee procedures were being met for the Unit 2 reactor vessel upper head:

- Reviewed the Effective Degradation Years and Reinspection Years calculations to determine if a volumetric examination or bare metal visual examination of the penetration nozzles was required during the current outage.
- Reviewed the examination report for the bare metal visual examination of a total of 83 penetrations were examined, including (78) CRDMs, (4) UHI and (1) Vent line. Verified that the examinations were performed in accordance with the requirements of the ASME Code and that the frequency was consistent with ASME Code Case N-729-4 and 10CFR50.55a dated July 18, 2017.
- Reviewed whether the required examination coverage was achieved and if limitations were recorded.

The inspectors verified that the licensee did not identify any indications that were accepted for continued service. Additionally, the inspectors verified that the licensee did not perform any welding repairs to the upper head penetrations since the beginning of commercial operation.

<u>Boric Acid Corrosion Control Inspection Activities</u>

The inspectors reviewed the licensee’s boric acid corrosion control program (BACCP) activities to determine if they were implemented in accordance with program requirements, applicable regulatory requirements, and industry guidance. Specifically, the inspectors performed the following activities:

- Reviewed applicable procedures and the results of the licensee’s most recent containment walkdown inspection.
- Verified that degraded or non-conforming conditions, such as boric acid leaks, were properly identified and corrected in accordance with the licensee’s BACCP and the CAP.
- Reviewed engineering evaluations of components with boric acid leakage which verified that minimum wall thickness of those components was maintained.

<u>Steam Generator Tube Inspection Activities</u>

The inspectors observed the following activities and/or reviewed the following documentation and evaluated them against the licensee’s technical specifications, commitments made to the NRC, ASME Section XI, and Nuclear Energy Institute 97-06 (Steam Generator Program Guidelines) and EPRI Guidelines:

- The inspectors assessed whether NDE flaw sizing accuracy was consistent with data from the Electric Power Research Institute (EPRI) examination technique specification sheets (ETSSs) by comparison of a sample of TVA-ETTSs used to inspect SG tubes, to the EPRI qualification ETTSs for the observed sample tube inspections
- Compared the numbers and sizes of SG tube flaws/degradation identified against the licensee’s previous Pre-Service Operational Assessment
- Reviewed the SG tube eddy current testing (ECT) examination scope and expansion criteria
- Evaluated if the licensee’s SG tube ECT examination scope including potential areas of tube degradation identified in the preservice SG tube inspections and/or as

identified in NRC generic industry operating experience applicable to the licensee's SG tubes
- Reviewed the licensee's repair criteria and processes, including the robot installation position verification system used during tube plugging activities
- Verified that primary-to-secondary leakage (e.g., SG tube leakage) was below three gallons per day, or the detection threshold, during the previous operating cycle according to licensee procedures
- Evaluated if the ECT equipment and techniques used by the licensee to acquire data from the SG tubes were qualified or validated to detect the known/expected types of SG tube degradation in accordance with Appendix "H" and "I" "Performance Demonstration for Eddy Current Examination, of EPRI Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 8
- Reviewed the WB-2 SG Degradation Assessment
- Observed qualified data analyst calibration setup and data analysis for a sample of tubes inspected within a tube sheet
- Observed the licensee's secondary side SG foreign object search and retrieval activities associated with SG "B"
- Observed and discussed resolution analysist dispositions for six steam generator tubes
- Reviewed a sample of ECT personnel qualifications
- Reviewed the Condition Monitoring and Operational Assessment Report tabulating the results and evaluation of the steam generator inspections performed during this (Fall 2017) refueling outage U2R1
- Review of in-situ test criteria with contract structural integrity analyst
- Observed FOSAR attempt on the secondary side of a tube sheet
- Observed FOSAR retrieval on primary side cross-under RCS piping
- Observed a sample of inspections performed of a primary side cladding inspection in accordance with Westinghouse NSAL-12-1 and Reviewed the SG-1 Channel Head Cladding Indication Assessment
- Conducted Interviews of TVA and contractor program leads as required to support above inspection areas

Identification and Resolution of Problems

The inspectors reviewed a sample of ISI-related issues entered into the corrective action program. The inspectors evaluated if the licensee had appropriately described the scope of the problem and had initiated corrective actions. The review also included the licensee's consideration and assessment of operating experience events applicable to the plant.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification and Performance (71111.11)

.1 Licensed Operator Requalification Review

a. Inspection Scope

On October 23, 2017, inspectors observed licensed operator just-in-time training being performed in preparation for the upcoming Unit 2 outage (i.e., evolution of placing one train of residual heat removal in service).

The inspectors specifically evaluated the following attributes related to the operating crews' performance:

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of operating procedures
- Control board operation and manipulation, including high-risk operator actions
- Command and Control provided by the unit supervisor and shift manager

This activity constituted one Observation of Requalification Activity inspection sample, as defined in IP 71111.11.

b. Findings

No findings were identified.

.2 Observation of Operator Performance

a. Inspection Scope

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 procedures 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. This activity constituted one Observation of Operator Performance inspection sample, as defined in IP 71111.11.

- 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.

.3 Annual Review of Licensee Requalification Examination Results

a. Inspection Scope

On October 19, 2017, the licensee completed the comprehensive biennial requalification written examinations and the annual requalification operating examinations required to be administered to all licensed operators in accordance with Title 10 of the *Code of Federal Regulations* 55.59(a)(2), “Requalification Requirements,” of the NRC’s “Operator’s Licenses.” The inspectors performed an in-office review of the overall pass/fail results of the individual operating examinations, written examinations, and the crew simulator operating examinations in accordance with IP 71111.11, “Licensed Operator Requalification Program.” These results were compared to the thresholds established in Section 3.02, “Requalification Examination Results,” of IP 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the performance-based problem 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) characterizing reliability issues for performance monitoring; 5) tracking unavailability for performance monitoring; 6) balancing reliability and unavailability; 7) trending key parameters for condition monitoring; 8) system classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); 9) appropriateness of performance criteria in accordance with 10 CFR 50.65(a)(2); and 10) appropriateness and adequacy of 10 CFR 50.65 (a)(1) goals, monitoring and corrective actions. This activity constituted four Maintenance Effectiveness inspection sample, as defined in IP 71111.12.

- CR 1329397, Loss of voltage to 6.9 kV shutdown board 1B
- Review of function 003B-B (auxiliary feedwater emergency supply) for 1B train of MDAFW
- WO 118279289, Cut and remove existing Unit 2 RWST half inch welded pipe and WO 118086469, 2-SI-61-2, 18 Month Ice Weighing (QC sample)
- Review of the Watts Bar Twelfth Periodic Summary Assessment Report (January 01, 2015 - May 31, 2016) 10CFR50.65 Paragraph (a)(3)

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors evaluated, as appropriate, the 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 activity constituted six Maintenance Risk Assessment inspection samples, as defined in IP 71111.13.

- Risk assessment for work week 1023 with Unit 1 SSPS repairs to number one steam generator Lo-Lo alarm while online
- Risk assessment for work week 1105 with component cooling system (CCS) 1A-A and MDAFW 1A-A out of service
- Risk assessment for October 27, 2017, with Unit 1 steam generator level channel troubleshooting and repair activities ongoing
- Risk assessment for October 30, 2017, with time to core boil at 25 minutes and containment closure required on Unit 2
- Review of Engineering Safety Feature room/area cooler maintenance risk impacts
- Risk assessment for November 6, 2017, with yellow shutdown risk on Unit 2

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the 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 (LCO) and the risk-significance in accordance with the SDP. The inspectors verified that the operability evaluations were performed in accordance with NPG-SPP-03.1, Corrective Action Program. This activity constituted three Operability Evaluation inspection samples, as defined in IP 71111.15.

- Prompt determination of operability (PDO) for CR 1357519, 2A train RHR minflow valve actuator cable damage
- PDO for CR 1228990, Pinhole leak on Unit 2 refueling water storage tank drain line
- Operability determination for the removal of CCS/AFW pump space cooler 1A from service

b. Findings

(Opened) Unresolved Item 05000390/2017004-01, Misapplication of Technical Specification Limiting Condition for Operation 3.0.6

Introduction. Inspectors identified an unresolved item (URI) associated with the misapplication of LCO 3.0.6 from the licensee's TS as it pertains to the functionality of engineering safety feature (ESF) coolers serving as non-TS support equipment. The item is unresolved pending the outcome of engineering analyses being performed by the licensee to determine if the ESF coolers are necessary for TS-supported systems to maintain operability.

Description. In April 2010, TVA revised the bases for the Watts Bar Unit 1 TS by adding language to expand the scope of LCO 3.0.6. The licensee evaluated the TS bases revision against the 10 CFR 50.59 criteria and determined a license amendment was not required for the change.

Prior to the TS bases revision, LCO 3.0.6 provided an exception for entering a supported system's conditions and required actions due to the inoperability of a TS support system which by definition is a support system that has an associated LCO in the TS. Following the TS bases revision, the scope of LCO 3.0.6 was expanded to allow another exception pertaining to non-TS support systems (i.e., support systems with no associated LCO) that are 100 percent redundant and have the capability of individually supporting both TS trains. Specifically, the revision allows both of the supported TS trains to be considered operable when one of the 100 percent redundant, non-TS support system trains is declared non-functional (i.e., the non-TS support systems do not have to meet the single failure criterion resulting in the TS systems not meeting the same criterion).

This revision to the TS bases manifests itself in the operation of both Unit 1 and Unit 2 ESF coolers that serve as 100 percent redundant, non-TS support systems for both trains of a TS system such as the emergency core cooling system (ECCS), containment spray (CS) system, and CCS. There are 12 plant areas with redundant trains of ESF coolers that support both trains of a TS system.

In support of maintenance and in accordance with operating procedure 0-SOI-30.05, TVA routinely removed one of the redundant coolers from service exposing the two trains of supported TS equipment to a single failure vulnerability. When the plant was in this configuration, the licensee considered both trains of the supported TS systems to be operable, and a TS LCO condition was not entered. The licensee justified operating in this manner based upon the interpretation of LCO 3.0.6, as previously discussed.

Inspectors reviewed control room logs and identified multiple occasions where the plant was operated in this manner. In response, the inspectors reviewed the licensing basis for Watts Bar and determined there was a discrepancy between: (1) the General Design

Criteria (GDC) found in Chapter 3 of the facility's UFSAR; and (2) the operation of TS systems with the single failure criterion not being met.

The GDC that pertain to ECCS, CS, and CCS all contain a requirement that the system safety function can be accomplished assuming a single failure. Inspection IMC 0326, "Operability Determination & Functionality Assessments for Conditions Adverse to Quality or Safety" (Agencywide Documents Access & Management System [ADAMS] Accession Number ML15328A099) contains guidance for inspectors to assist their review of licensee determination of operability and resolution of degraded or nonconforming conditions. IMC 0326 specifies that failure to meet a GDC is an entry point for an operability determination. Also, based on the definition of operability, IMC 0326 states: "The operability requirements for an SSC [structure, system, and component] encompass all necessary support systems (per the TS definition of operability) regardless of whether the TS explicitly specify operability requirements for the support functions."

In response to this inspection discovery, the licensee took two actions. First, in the nearterm, TVA revised the applicable ESF cooler operating procedure to require entrance into the appropriate LCO condition and required action statement when one of the ESF cooler trains is nonfunctional. Secondly, while the current design bases for the affected systems indicates that the coolers are required for system operability, TVA is performing engineering evaluations to determine if the support requirement can be eliminated under certain conditions. This effort is being tracked in TVA's corrective action program by CR 1357258. Based on the ongoing engineering evaluations, the inspectors have characterized this issue as a URI pending the outcome of the results. Once the evaluations are finalized, additional inspection can be performed to determine if a PD actually exists (e.g., TS violation). This is identified as URI 05000390/2017004-01, Misapplication of Technical Specification Limiting Condition for Operation 3.0.6.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the permanent plant modifications listed below 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. This activity constituted two Plant Modifications inspection samples, as defined in IP 71111.18.

- Review of Engineering Document Change (EDC) 54175, for change to limiting condition for operation 3.0.6 in the Unit 1 TS Bases
- Review of CR 1352679, Temporary pump used for completing transfer canal drain

b. Findings

.1 See URI 2017-004-01 documented in Section 1R15 of this report.

.2 Inadequate Procedure for Temporary Configuration Changes, NCV 05000390, 391/2017004-02

Introduction. A self-revealed Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for inadequacies associated with TVA procedure NPG-SPP-09.5, Temporary Modifications Temporary Configuration Changes, Revision 11. Specifically, a procedural exception allowed a temporary configuration change to be installed in the spent fuel pool without a screening in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments." The change subsequently caused an inadvertent draining of the level of the spent fuel pool to the point the control room received the low level alarm.

Description. On October 28, 2017, at approximately 11:19 a.m., site personnel installed a temporary pump and hose in order to complete planned draining of the spent fuel pool transfer canal. The pump/hose was configured to pump water from the transfer canal into the common spent fuel pool. During the evolution a condition developed that resulted in a siphon effect that began draining the spent fuel pool inventory back into the transfer canal. This continued until the control room received the low level alarm in the spent fuel pool and dispatched personnel to the area. After responding to the event, personnel stopped the inadvertent draining of the spent fuel pool. During this event, Watts Bar Unit 1 was at 100 percent power. Watts Bar Unit 2 was reducing power for a refueling outage. Unit 2 had entered mode 4 on October 28, 2017, at 11:05 am, and entered mode 5 at 4:16 p.m. on the same day. Inspectors reviewed TVA Temporary Modifications Temporary Configuration Changes, NPG-SPP-09.5, Revision 11. Section 3.2.2, "Exclusions," Step "M" of this procedure states:

"The use of auxiliary equipment for filling or draining system components (for example tanks or pools) as long as the filling or draining activity is not credited with maintaining component/system operability, and the auxiliary equipment is removed when the evolution is complete."

Inspectors determined that this procedure exclusion for an otherwise required screening under 10 CFR 50.59 did not address the potential for these types of filling/draining activities to create an operability challenge due to equipment failure or human performance errors. Instead, the exclusion is only focused on activities that are credited with maintaining operability. Inspectors determined that the events on October 28, 2017, were further complicated by the following:

- Personnel did not use any process or document, such as a WO, to control the installation, use, or removal of the temporary equipment.
- Personnel did not perform a review in accordance with 10CFR 50.59, "Changes, Tests, and Experiments" for the use of the temporary pump.
- Main control room narrative log entries were inadequate because they lacked any details associated with the event beyond the receipt of the Hi/Low level alarm itself.
- No CR was written that described the condition adverse to quality associated with the event or sufficient details of what occurred.

The licensee entered all of the above issues into their corrective action program through CRs 1354018, 1354479, 1354552, 1354684, and 1357392. Inspectors determined that the volume of the transfer canal was such that the TS limit for spent fuel pool level would have been exceeded if no operator action was taken.

Analysis. The failure to ensure procedure NPG-SPP-09.5, Temporary Modifications Temporary Configuration Changes, Revision 11, was appropriate to the circumstances was a performance deficiency. Specifically, Step 3.2.2.M inappropriately allowed an exclusion from review in accordance with this procedure and the 10 CFR 50.59 screening/evaluation process for an activity that involved filling the spent fuel pool with a temporary pump on October 28, 2017. As a result, the use of the temporary pump caused the inadvertent draining of the spent fuel pool to the point that the low level alarm was received in the main control room. The performance deficiency was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of design control, and it adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors performed the significance determination using NRC IMC 0609. The finding affected the common spent fuel pool and the Barrier Integrity Cornerstone while Unit 1 was at power and Unit 2 was in mode 4. Therefore, IMC 0609.04, “Initial Characterization of Findings,” dated October 7, 2016, was used for Unit 1 and IMC 0609 Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings,” dated May 9, 2014, was used for Unit 2. Using IMC 0609 Appendix A, Exhibit 3 – Barrier Integrity Screening Questions, for Unit 1, the inspectors determined that this finding was of very low safety significance (Green) because the finding did not result in a loss of spent fuel pool inventory below the minimum analyzed level in the site-specific licensing basis. Using IMC 0609 Appendix G, Attachment 1, Exhibit 4 - Barrier Integrity Screening Questions, for Unit 2, the inspectors determined that this finding was of very low safety significance (Green) because it did not involve an actual reduction in function of hydrogen control for PWR ice condenser containments.

The finding had a cross-cutting aspect in the Work Management attribute of the Human Performance area as defined in NRC IMC 0310, because the licensee did not implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. (H.5)

Enforcement. Title 10 CFR Part 50, Appendix B to 10 CFR Part 50, 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.” This requirement is implemented, in part, by TVA procedure NPG-SPP-09.5, “Temporary Modifications Temporary Configuration Changes,” Revision 11. Contrary to the above, on October 28, 2017, NPG-SPP-09.5 was not appropriate to the circumstances because it allowed the installation and use of a temporary pump connected to the spent fuel pool, which had the capability to inadvertently drain a portion of the spent fuel pool, without a requirement to perform a 10 CFR 50.59 screening for the change. The licensee completed immediate corrective actions to stop the inadvertent siphon and arrest the spent fuel pool level loss.

This violation is being treated as an NCV consistent with section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s corrective action program as CR 1354018. This violation is identified as NCV 05000390, 391/2017004-02, Inadequate Procedure for Temporary Configuration Changes.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the 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. This activity constituted five Post Maintenance Testing inspection samples, as defined in IP 71111.19.

- WO 119145005, 1-RLY-99-K230-B, Post maintenance test of the Unit 1 SSPS K230 relay, which supports steam generator 1 low-low level reactor trip function
- WO119219436, 2-SI-268-1-A, 92 Day permanent hydrogen mitigation system train A igniter availability test following igniter repairs
- WO119219435, 2-SI-268-1-B, 92 Day permanent hydrogen mitigation system train B igniter availability test following igniter repairs
- WO 118085986, 2-SI-99-10B, Unit 2 SSPS logic testing issues during testing (CR 1355656)
- WO 118279289, Qualified visual inspection of weld repair to half inch refueling water storage tank drain line

b. Findings

No findings were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Refueling Outage Cycle 1

a. Inspection Scope

The inspectors reviewed the outage risk control plan for the Unit 2 Cycle 1 (2RFO1) refueling outage (RFO) to assess whether the licensee had appropriately considered risk, industry experience, and previous site-specific problems, and to also confirm that the licensee had mitigation/response strategies for losses of key safety functions.

The licensee began its 2RFO1 refueling outage on October 28, 2017, and ended the outage on December 6, 2017. During that period, the inspectors observed portions of the shutdown, cooldown, fueling, and maintenance activities to verify that the licensee maintained defense-in-depth (DID) commensurate with the outage risk plan and applicable TS.

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.

- Licensee configuration management, including daily outage reports, to evaluate DID commensurate with the outage safety plan and compliance with the applicable TS when taking equipment out of service.
- Installation and configuration of reactor coolant instruments to provide accurate indication and an accounting for instrument error
- Controls over the status and configuration of electrical systems and switchyard to ensure that TS and outage safety plan requirements were met
- Decay heat removal processes to verify proper operation and that steam generators, when relied upon, were a viable means of backup cooling
- Controls to ensure that outage work was not impacting the ability to operate the spent fuel pool cooling system during and after core offload
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss
- Reactivity controls to verify compliance with TS and to verify that activities which could affect reactivity were reviewed for proper control within the outage risk plan

b. Findings

Failure to Promptly Identify a Condition Adverse to Quality for a Boric Acid Leak on 2-SMV-68-548, NCV 05000391/2017004-03

Introduction. A NRC-identified NCV of very low safety significance (Green) associated with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to promptly identify and correct a condition adverse to quality. Specifically, NRC inspectors identified a boric acid leak on the Unit 2 loop 1 hot leg sample valve, 2-SMV-68-548, that had been missed by licensee personnel performing boric acid corrosion control program walkdowns during Unit 2 refueling outage 1 (RFO1).

Description. On November 28, 2017, inspectors were performing the containment closeout inspection of lower containment on Unit 2 in accordance with inspection procedure 71111.20, "Refueling and Other Outage Activities". During the walkdown, inspectors identified a boric acid leak on 2-SMV-68-548, the Unit 2 loop 1 hot leg sample valve. The boric acid was dry at the time, but evidence of a wet leak during operation and discoloration were present. At the time of the walkdown, the unit was in mode 5 with significantly lower temperatures and pressures than during normal operation (mode 1). The licensee subsequently determined that this leak had been missed during all previous boric acid control programs and other routine containment walkdowns during RFO1. The licensee also determined that the valve was leaking through the packing along the yoke to body interface, (i.e., not a pressure boundary or through wall leak). A review of the valve properties revealed that the KeroTest ¾-inch globe valve had a carbon steel yoke and several other internal parts that were conservatively assumed to be carbon steel based on best available information. As such, the potential for further degradation due to boric acid was possible if the condition was not corrected. The valve is located at the 12 o'clock position on the loop 1 hot leg prior to the #1 steam generator. The section of lower containment where the valve is located is not accessible at power. The licensee cleaned up the boric acid and decided to implement a temporary modification to close the manual valve to isolate the leak pathway, since it was not required for operation during the cycle.

The inspectors reviewed the issue and determined that the licensee was required to promptly identify and correct the condition adverse to quality associated with the boric acid leak on 2-SMV-68-548 in accordance with 10 CFR 50, Appendix B, Criterion XVI. The licensee wrote CR 1362880 to document the leakage and CR 1362384 to drive improvements in boric acid control program containment walkdowns during outages.

Analysis. The failure to promptly identify and correct a condition adverse to quality associated with a boric acid leak on 2-SMV-68-548 as required by 10 CFR Part 50 Criterion XVI, "Corrective Action," was a performance deficiency. The performance deficiency was determined to be more than minor because, if left uncorrected, it had the potential to lead to a more significant safety concern. The inspectors performed the significance determination using NRC IMC 0609. The finding affected the Barrier Integrity Cornerstone while Unit 2 was shut down, so IMC 0609 Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings," dated May 9, 2014, was used to determine that this finding was of very low safety significance (Green) because it did not degrade the ability to isolate a drain down or leakage path.

The finding had a cross-cutting aspect in the Work Management attribute of the Human Performance area as defined in NRC IMC 0310, because the licensee did not implement a process of planning, controlling, and executing work activities such that nuclear safety was the overriding priority. (H.5)

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that "measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Contrary to the above, on November 28, 2017, a condition adverse to quality was not promptly identified and corrected. Specifically, the licensee did not identify a condition adverse to quality associated with a boric acid leak on Unit 2 loop 1 hot leg sample valve 2-SMV-68-548 and enter it into their corrective action program. The licensee's immediate corrective actions included implementing a temporary modification to allow the valve to be closed to isolate the leakage. This violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's corrective action program as CR 1362880. This violation is identified as NCV 05000391/2017004-03, Failure to Promptly Identify a Condition Adverse to Quality for a Boric Acid Leak on 2-SMV-68-548.

.2 Unit 2 Forced Outage (December 11, 2017 – December 15, 2017)

a. Inspection Scope

On December 11, 2017, Unit 2 was manually tripped after two control rods fell into the core during stable 100 percent power operations. The inspectors responded to the control room and observed the immediate operator and plant response to the trip. Inspectors subsequently observed portions of the plant startup including reactor criticality and power ascension. The inspectors verified that mode changes were performed in accordance with plant TS. This inspection satisfied one inspection sample for Outage Activities, as defined in IP 71111.20.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed the 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. This activity constituted seven Surveillance Testing inspection samples; two in-service; one routine; three ice condenser; and one containment isolation valve, as defined in IP 71111.22.

In-Service Test:
- WO 118645474, 2-SI-72-904-A, Check valve testing during operations (train A)
- WO 118086271, 2-SI-63-905, Boron injection check valve flow test during refuel outages

Other Surveillances
- WO 118086645, 0-SI-82-5, 18 month loss of offsite power with safety injection, EDG 2A-A

Ice Condenser
- WO 118863944, 18 month lower inlet door inspection
- WO 118376713, 18 month intermediate deck door inspection
- WO 118086469, Ice weighing

Containment Isolation Valve
- Unit 2 local leak rate testing of penetration X-34

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

.1 Cornerstone: Mitigating Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the six PIs listed below. To verify the accuracy of the PI data reported from July 1, 2016 through June 30, 2017. PI definitions and guidance contained in NEI 99-02, Regulatory Assessment Indicator Guideline, Revision 7, were used to verify the basis in reporting for each data element. This activity constituted six performance indicator samples, as defined in IP 71151.

- Unit 2, Reactor coolant system activity
- Unit 2, Unplanned scrams with complications
- Unit 2, RCS leak rate
- Unit 2, Safety system functional failures
- Unit 1, Safety system functional failures
- Unit 1, Cooling water MSPI

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, Problem Identification and Resolution, 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 corrective action program. This review was accomplished by reviewing daily condition report (CR) summary reports and attending daily CR review meetings

.2 Annual Sample: CR 1324531, Unit 1 Auxiliary feedwater (AFW) pressure control valve failed during surveillance testing

a. Inspection Scope

The inspectors conducted a detailed review of CR 1324531, and the sample was selected based on maintenance rule insights. The inspectors evaluated the following attributes of the licensee's actions:

- complete and accurate identification of the problem in a timely manner;
- evaluation and timely disposition of operability and reportability issues;
- consideration of extent of condition, generic implications, common cause, and previous occurrences;
- classification and prioritization of the problem;

- identification of root and contributing causes;
- identification of effective corrective actions; and
- completion of corrective actions in a timely manner

b. Findings

No findings were identified. The inspectors determined that:

- CR 1324531 was of sufficient detail and initiated in a timely manner;
- operability evaluation was adequate;
- consideration of extent of condition, generic implications, and common cause were appropriate;
- cause identification was adequate; and
- corrective actions were appropriate and completed or scheduled in a timely manner.

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report (LER) 05000390/2016-002-00, Technical Specification Action Not Met for Inoperable Containment Isolation Valve

a. Inspection Scope

This LER discussed an improper interpretation of Technical Specification 3.6.3, "Containment Isolation Valves." The licensee stated in the LER that Watts Bar Nuclear Plant Unit 1 entered TS 3.6.3, Condition A, for an inoperability of a containment isolation valve. The licensee discovered leakage on valve 1-FCV-61-122, Glycol cooled floor return header isolation and, subsequently, declared this valve inoperable. TS 3.6.3, Condition A, requires that a penetration flow path with one containment isolation valve inoperable to be isolated by use of at least one closed and de-automatic valve, closed manual valve, blind flange, or check valve with flow through the vale within four hours. In this particular event, the licensee did not isolate the penetration associated with this containment isolation valve until six hours after the event. The inspectors reviewed the LER associated with this event and determined that the report adequately documented the summary of the event including the cause of the event and potential safety consequences.

b. Findings

The enforcement aspects of this violation are discussed in Section 4OA7.

.2 (Closed) Licensee Event Report (LER) 05000391/2017-001-00, Containment Airlock Function Lost Due to Equalizing Valve Not Closing

a. Inspection Scope

The inspectors reviewed LER 05000391/2017-001-00, dated May 3, 2017. This LER discussed the loss of the containment airlock function when the inboard equalizing valve was found open with the outer airlock door open. This created a containment bypass with leakage.

On March 9, 2017 at 0120 hours Eastern Standard Time (EST), the licensee determined that the pressure equalizing valve for the Watts Bar Nuclear Plant, Unit 2 upper containment airlock inboard door was found not closed while the outboard airlock door was open. As such, TS Section 3.6.2, "Containment Air Locks," Condition C, and TS Section 3.6.1, "Containment," were not met. The outer door was shut to isolate the airlock on 0125 hours EST. The inner door was then cycled which closed the equalizing valve. This action allowed the licensee to exit TS LCOs 3.6.2 Condition C and TS LCO 3.6.1. The outer door was opened and the licensee verified there was no leakage to the containment. The total time that a containment bypass was present was estimated to be five minutes.

This event was entered into the TVA corrective action program and is being tracked under CR 1270608. The licensee discovered the leaking equalizing valve, the inner door was cycled and the valve was seated. The licensee repaired the valve by replacing the shoulder bolt and performing a post maintenance test.

The NRC staff reviewed the LER to verify that it met the guidance described in NUREG-1022, 'Event Report Guidelines 10 CFR 50.72 and 50.73,' Revision 3. The NRC staff reviewed CR 127608. This LER is closed.

b. Findings

No findings were identified.

.3 (Closed) Licensee Event Report (LER) 05000391/2017-002-00, Manual Reactor Trip as a Result of a Secondary Plant Transient

a. Inspection Scope

This LER discusses a manual reactor trip initiated on Unit 2 due to lowering steam generator levels on March 20, 2017. The lowering steam generator levels were caused by an inadvertent trip of the 2A hotwell Pump. The 2A hotwell pump trip was a result of scaffold workers inadvertently depressing the local trip pushbutton. The tripping of the 2A hotwell pump caused an automatic trip of the 2C condensate booster pump four minutes later. Operations personnel attempted to lower reactor power from100 percent RTP utilizing 2-AOI-29, Rapid Load Reduction. Power was reduced to 91 percent RTP. At this point, steam generator levels were recovering. Approximately five minutes after the start of the load reduction, the 2B condensate booster pump tripped due to low suction pressure. With the loss of the second condensate booster pump, steam generator levels began to lower. Due to the loss of multiple secondary pumps, operations personnel determined that recovery of steam generator water level was not possible and inserted a manual reactor trip prior to reaching an automatic trip setpoint.

The inspectors reviewed the LER and trip report associated with this event and determined that the report adequately documented the summary of the event including the cause of the event and potential safety consequences. The inspectors also reviewed the corrective actions and found they were adequate for cause of the event. The local trip switches on several secondary pumps were protected with bump guards to prevent inadvertent operation. In addition training was conducted with operations and construction personnel on working in trip sensitive areas. All these corrective actions were completed at the time of this inspection report.

b. Findings

No findings were identified.

.4 (Closed) LER 05000391/2017-003-00, Automatic Start of Auxiliary Feedwater System Due to Main Condenser Failure

a. Inspection Scope

On March 23, 2017, at 0014 Eastern Daylight Time (EDT) with the plant at 18 percent power, WBN Unit 2 experienced an unplanned trip of both turbine driven main feed pumps (TDMFPs) following a loss of main condenser vacuum. The trip of both TDMFPs caused an automatic start of the auxiliary feedwater (AFW) system, including both MDAFW pumps and the TDAFW pump as designed. A breach on the B zone of the main condenser caused the loss of main condenser vacuum. Investigation determined that inadequate structural support at the neck of the main condenser caused the breach, which resulted in the cracking and ultimately the failure of the gusset plates. Since the event resulted in the automatic actuation of the AFW system, which is one of the systems listed under 10 CFR 50.73(a)(2)(iv)B, then the event was reportable per 10 CFR 50.73(a)(2)(iv)(A) due to the automatic operation of a system listed in 10 CFR 50.73(a)(2)(iv)(B). The inspectors reviewed the licensee's root cause analysis (RCA) associated with CR 1275870. The review included the licensee's analysis of the event, their determination of the direct and contributing causes of the event, and the licensee's corrective actions to address the cause of the event. The inspectors determined that the licensee's investigation was thorough and the corrective actions that have been completed and those planned for implementation appropriately address the identified causes of the event. This LER is closed.

b. Findings

No findings were identified.

.5 (Closed) LER 05000390/2016-005-01, Both Trains of Unit 1 Emergency Gas Treatment System Inoperable During Unit 2 Testing

a. Inspection Scope

LER 05000390/2016-005-00 is described and closed in Watts Bar inspection report 05000390, 391/2017-003. The original LER was supplemented on July 13, 2016, with LER 05000390/2016-005-01. The supplement updated the results of the safety system functional failure consideration to conclude that the event constituted a safety system functional failure for the emergency gas treatment system in accordance with NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 7. This LER was reviewed by the inspectors.

b. Findings

No findings were identified.

.6 (Closed) LER 05000390/2016-011-01, Loss of Centrifugal Charging Pump Due to Repeat Failure of Associated Room Cooler

a. Inspection Scope

LER 05000390/2016-011-00 is described and closed in Watts Bar inspection report 05000390, 391/2017-003. The original LER was supplemented on December 9, 2016, with LER 05000390/2016-011-01. The supplement provided additional information about the results of the cause evaluation, associated corrective actions, the safety consequences of the event, and the safety system functional failure determination. This LER was reviewed by the inspectors.

b. Findings

No findings were identified.

.7 (Closed) LERs 05000391/2016-003-00, 01, Turbine Driven Auxiliary Feedwater Pump Inoperable for Longer than Allowable Outage Time due to Governor Spring Over-Tensioning

a. Inspection Scope

The Unit 2 TDAFW pump auto-started upon a planned reactor trip at 0154 EDT on May 28, 2016. At 0157 EDT the reactor operator noted that TDAFW forward flow to steam generators 1 and 3 were approximately 800 gallons per minute and placed the associated level control valves in the closed position. At approximately 0203 EDT the main control room received an alarm for the TDAFW pump electrical overspeed trip. Operators walked down the TDAFW pump and determined that the turbine had tripped, by confirming that the trip and throttle valve was no longer latched, and declared the TDAFW pump inoperable. The equipment was repaired, the TDAFW pump was retested successfully and returned to service. TS LCO 3.7.5 was exited on May 30, 2016. The cause of the failure was determined to be an over-tensioned stem spring on the governor valve for the pump. The original LER was supplemented on September 2, 2016, with LER 391/2016-003-01. The supplement provided additional information about the results of the cause evaluation, associated corrective actions, and the safety system functional failure determination. The inspectors reviewed the licensee's apparent cause evaluation associated with CR 1175968. The review included the licensee's analysis of the event, their determination of the direct and contributing causes of the event, and the licensee's corrective actions to address the cause of the event. The inspectors determined that the licensee's investigation was thorough and the corrective actions that have been completed and those planned for implementation appropriately address the identified causes of the event. An NCV for this condition was documented in NRC Inspection Report 05000390, 391/2016-002-08. These LERs are closed.

b. Findings

No findings were identified.

.8 Unit 2 Manual Reactor trip due to Multiple Dropped Rods

a. Inspection Scope

The inspectors responded to a Unit 2 manual reactor trip that occurred on December 11, 2017, due to four control rods spontaneously dropping ointo the core. At 0857 EST on Decemebr 11, 2017, while operating at 97 percent power, all four control rods in control bank A, group 2 dropped into the core. The reactor operator, after verifying by diverse indication that the rods had actually dropped, manually tripped the reactor as required by procedure when more than one rod has dropped.

The inspectors discussed the preliminary cause of the trip with the licensee and reviewed unit parameters and system response to verify that equipment responded to the reactor trip as designed. The inspectors also reviewed the licensee's post-trip report. The inspectors reviewed the event notification to verify that it met regulatory requirements.

b. Findings

No findings were identified.

4OA6 Meetings, including Exit

On January 19, 2018, the resident inspectors presented the quarterly inspection results to members of the licensee staff. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

On November 15, 2017, the inspectors presented the inspection results to Mr. T. Marshall, Plant Manager, and other members of the licensee staff. The inspectors confirmed that all proprietary information reviewed during the inspection was returned and that none of the potential report input discussed was considered proprietary.

4OA7 Licensee-Identified Violations

The following licensee-identified violation of NRC requirements was determined to be of very low safety significance and met the NRC Enforcement Policy criteria for being dispositioned as a Non-Cited violation.

Watts Bar Unit 1 TS 3.6.3, "Containment isolation Valve," Condition A, states, in part, that a penetration flow path with one containment isolation valve inoperable to be isolated by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured within four hours. Contrary to this requirement, Watts Bar Nuclear Plant Unit 1 containment isolation valve 1-FCV-61-122, Glycol cooled floor return header isolation was inoperable on March 5, 2016, at 1512 EST, and the penetration associated with this containment isolation valve was not isolated until 2113 EST on March 5, 2016. TS 3.6.3, Condition A, has a required completion time of four hours; however, valve 1-FCV-61-110 was not closed until six hours into the event. The licensee has entered this event into their corrective action program under CR 1146157. This licensee-identified violation of NRC requirements was determined to be of very low safety significance, Severity Level IV, and met the NRC Enforcement Policy Section 2.3.2 criteria for being dispositioned as a non-cited violation. The performance deficiency was more than minor because it was associated with the reactor containment barrier performance attribute of the barrier cornerstone in NRC IMC 0609, Attachment 04, dated October 7, 2016. This finding was further evaluated in accordance with NRC IMC 0609, Appendix A, The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012. The finding screened to Green since there was no actual open pathway in the physical integrity of the reactor containment.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Arent, Assistant to the VP
M. Casner, Operations Director
L. Cross, Manager, Electrical Systems
T. Detchemendy, Manager, Site Emergency Preparedness
E. Ellis, Senior Manager, Nuclear Site Security
K. Hulvey, Watts Bar Licensing Manager
J. James, Director, Maintenance
B. Jenkins, Director, Engineering
T. Marshall, Plant Manager
C. Rice, Operations Superintendent
P. Simmons, Site Vice President
A. White, Senior Manager, Site Quality Assurance

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000391/2017004-03	NCV	Failure to Promptly Identify a Condition Adverse to Quality for a Boric Acid Leak on 2-SMV-68-548 (Section 1R20.1)
05000390, 391/2017004-02	NCV	Inadequate Procedure for Temporary Configuration Changes (Section 1R18.2)

Opened

05000390/2017004-01	URI	Misapplication of Technical Specification Limiting Condition for Operation 3.0.6 (Section 1R15)

Closed

05000390/2016-002-00	LER	Technical Specification Action Not Met for Inoperable Containment Isolation Valve (Section 4OA3.1)
05000391/2017-001-00	LER	Containment Airlock Function Lost Due to Equalizing Valve Not Closing (Section 4OA3.2)
05000391/2017-002-00	LER	Manual Reactor Trip as a Result of a Secondary Plant Transient (Section 4OA3.3)
05000391/2017-003-00	LER	Automatic Start of Auxiliary Feedwater System Due to Main Condenser Failure (Section 4OA3.4)
05000390/2016-005-01	LER	Both Trains of Unit 1 Emergency Gas Treatment System Inoperable During Unit 2 Testing (Section 4OA3.5)
05000390/2016-011-01	LER	Loss of Centrifugal Charging Pump Due to Repeat Failure of Associated Room Cooler (Section 4OA3.6)
05000391/2016-003-00, 01	LER	Turbine Driven Auxiliary Feedwater Pump Inoperable for Longer than Allowable Outage Time due to Governor Spring Over-Tensioning (Section 4OA3.7)

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0-PI-OPS-1-FP, Freeze Protection, Rev. 0015
NPG-SPP-07.1.7, Station Seasonal Readiness, Rev. 0005
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CR 1234334
Certification of 2017 Winter Readiness – Watts Bar

Section 1R04: Equipment Alignment

Procedures

0-PI-OPS-17.0, 18 Month Locked Valve Verification, Rev. 0082
0-PI-OPS-17.1, 18 Month Locked Breaker Verification, Rev. 0026
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RHR Power Checklist 2-74.01-3P; 2-SOI-74.01 Attachment – 3P, Rev. 0006, effective date 03/13/16
RHR Power Checklist 2-74.01-1P; 2-SOI-74.01 Attachment – 1P, Rev. 0008, effective date 06/23/16
0-PI-OPS-17.0, 18 Month Locked Valve Verification, Attachment 5, Unit 2 Valve Verification, and Rev. 0082
0-PI-OPS-17.0, 18 Month Locked Valve Verification, Attachment 6, Unit 2 Valve Verification, Rev. 0082
Drawing 2-47W810-1, Rev. 23
2-SOI-74.01 ATT-1V, Residual Heat Removal System, Valve Checklist 2-74.01-1V, Rev. 0000
2-SOI-74.01 ATT-2P, Residual Heat Removal System, Power Checklist 2-74.01-2P, Rev. 0000
2-SOI-74.01 ATT-2V, Residual Heat Removal System, Valve Checklist 2-74.01-2V, Rev. 0000
2-SOI-74.01 ATT-3V, Residual Heat Removal System, Valve Checklist 2-74.01-3V, Rev. 0000
2-SOI-74.01 ATT-3P, Residual Heat Removal System, Power Checklist 2-74.01-3P, Rev. 0006
2-SOI-74.01 ATT-1P, Residual Heat Removal System, Power Checklist 2-74.01-1P, Rev. 0008
2-SOI-72.01 ATT 1V, Containment Spray System, Valve Checklist 2-72.01-1V, Rev. 0001
2-SOI-63.01 ATT 1V, Safety Injection System, Valve Checklist 2-63.01-1V, Rev. 0005
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1-SOI-3.02 ATT 1V, Auxiliary Feedwater System, Valve Checklist 103.02-1V, Rev. 0014
1-SI-3-80, 18 Month Channel Calibration of Auxiliary Feedwater Pump 1A-A Differenetial Pressure Loops 1-LPP-3-122C and 1-LPP-3-122A, Rev. 0019
1-SOI-3.02 Att 1H, Auxiliary Feedwater System Handswitch Checklist 1-3.02-1H, Rev. 0008
1-SOI-3.02 ATT 1P, Auxiliary Feedwater System Power Checklist 1-3.02-1P, Rev. 0013
1-SOI-72.01, Containment Spray Ssytem, Rev. 0003
0-PI-OPS-17.0, 18 Month Locked Valve Verification, Rev. 0084
1-SI-72-701, Containment Spray and RHR Spray Header Valve Water Inventory Test, Rev. 0014
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WBN-Prefire Plan, AUX-0-692-02, Rev. 3
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Drawing 47A381-20
Drawing 47A381-127
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WB-DC-40-31.51, Evaluating the Effects of Flooding Due to Moderate Energy Pipe Failures Inside and Outside Containment
CEB Report 77-55, Evaluation of the Effects of Postulated Pipe Failures Outside of Containment for Watts Bar Nuclear Plant, Units 1 and 2
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SGP-PLG-RIB-FP-GEN, Westinghouse Mechanical Ribbed Plugging of Steam Generator Tubes, Rev 4.1
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