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

November 22, 2017

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 – NUCLEAR REGULATORY COMMISSION INTEGRATED INSPECTION REPORT 05000390/2017003, 05000391/2017003

Dear Mr. Shea:

On September 30, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Watts Bar Nuclear Plant, Unit 1 and Unit 2. On October 25, 2017, the NRC inspectors discussed the results of this inspection with Mr. Tom Marshall and other members of your staff. A re-exit was conducted on November 8, 2017, with Ms. Kim Hulvey. The results of this inspection are documented in the enclosed inspection report.

The NRC inspectors documented three findings of very low safety significance (Green) in this report which also involved violations of NRC requirements. Additionally, inspectors documented six licensee-identified violations which were determined to be of very low safety significance in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy. If you contest these 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/

Alan Blamey, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos.: 50-390, 50-391
License Nos.: NPF-90, 96

Enclosure:
IR 05000390/2017003, 05000391/2017003
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.:	05000390/2017003, 05000391/2017003
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Units 1 and 2
Location:	Spring City, TN 37381
Dates:	July 1 through September 30, 2017
Inspectors:	J. Nadel, Senior Resident Inspector J. Hamman, Resident Inspector J. Jandovitz, Senior Resident Inspector E. Lea, Regional Government Liaison Officer S. Freeman, Senior Reactor Analyst J. Eargle, Senior Construction Inspector B. Bishop, Project Engineer G. Crespo, Senior Construction Inspector C. Rapp, Senior Project Engineer R. Taylor, Senior Project Inspector B. Davis, Senior Construction Inspector
Approved by:	Alan Blamey, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY

IR 05000390/2017-003; 05000391/2017-003; July 1, 2017 – September 30, 2017; Watts Bar Nuclear Plant; Operability Evaluations, Surveillance Testing.

The report covered a three-month period of inspection by the resident inspectors. Three Green non-cited violations (NCV) 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: Mitigating Systems

- Green. An NRC-identified NCV was identified for the failure to maintain written procedures for emergencies. Emergency procedure 1-E-1, Revision 7 and 2-E-1 Revision 0, both titled Loss of Reactor or Secondary Coolant, were updated to include steps directing inappropriate actions that would have affected emergency raw cooling water (ERCW) supply flow during an accident. The immediate corrective action was to remove the inappropriate steps. This violation was documented in the licensee's corrective action program (CAP) as CR 1331422.

  The performance deficiency was more than minor because it affected the Mitigating Systems Cornerstone attribute of Procedure Quality and adversely affected the cornerstone objective in that the reduced ERCW flow caused by the inappropriate steps affects the heat removal capability of the ERCW and component cooling systems (CCS) during a loss of coolant accident (LOCA). The finding was determined to require a detailed risk evaluation because it represented an actual loss of function of at least a single train for greater than its TS allowed outage time. The result was less than 1E-6 for each unit which would be a finding of very low significance (Green). The risk was mitigated because the performance deficiency would affect operation only when a LOCA occurred and simultaneous loss of two shutdown boards. The finding has a cross-cutting aspect in the documentation attribute of the Human Performance area because the licensee did not maintain the accuracy of 1-E-1 through its revisions and did not maintain procedure 2-E-1 accurate at its creation. (H.7) (Section 1R15)

- Green. An NRC-identified NCV of Technical Specification (TS) 5.7.1.1.a, "Procedures," was identified for the failure to maintain TVA procedures 1-GO-6 and 2-GO-6, both titled Unit Shutdown from Hot Standby to Cold Shutdown. The licensee failed to update the procedures prior to commencing dual unit operation to include steps that would shut down the running motor driven auxiliary feedwater pump prior to starting a third ERCW pump during the time period where the opposite unit has been shut down less than 48 hours. The licensee's immediate corrective actions included revising both procedures to add the required steps. This violation was documented in the licensee's CAP as CR 1318176.

The performance deficiency was more than minor because it affected the Mitigating Systems Cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective in that failure to maintain the procedures resulted in a situation where the emergency diesel generator would have been rendered inoperable during a design basis event. The inspectors determined the finding was of very low safety significance (Green) because the finding did not represent an actual loss of function of a single train for greater than its TS allowed outage time. The finding had a cross-cutting aspect in the Avoid Complacency attribute of the Human Performance area because engineering missed a critical aspect of the required procedure changes associated with design change notice 62151 when performing the prompt determination of operability and the review process was unsuccessful at identifying the error [H.12]. (Section 1R15)

Cornerstone: Initiating Events

- Green. A self-revealed NCV of (TS) 5.7.1.1.a, "Procedures," was identified for the failure to follow TVA procedure 2-SI-68-86, 18 month Channel Calibration of Remote Shutdown Monitoring Narrow Range Pressurizer Pressure Loop 2-LPP-68-337C, revision 4. The licensee failed to properly follow step 6.2.6 [1.3], which resulted in the inadvertent lifting of a pressurizer power operated relief valve (PORV). The licensee's immediate corrective actions included revising the procedure. This violation was documented in the licensee's CAP as CR 1309345.

  The performance deficiency was more than minor because it affected the Initiating Events Cornerstone attribute of Human Performance and adversely affected the cornerstone objective in that failing to follow procedure 2-SI-68-86 caused a depressurization of the plant that had to be stopped by operator action. The finding was determined to be very low safety significance (Green) because the resultant leakage from the open PORV would be self-limiting such that it would stop before impacting the operating method of decay heat removal. The finding had a cross-cutting aspect in the Challenge the Unknown component of the Human Performance area as defined in NRC IMC 0310, because the technicians failed to recognize that the output was already set to 0, but proceeded anyway to toggle the output which resulted in setting it to 1 [H.11]. (Section 1R22)

Six violations of very low safety significance, identified by the licensee, have been reviewed by the NRC. Corrective actions taken or planned by the licensee have been entered into the licensee's CAP. These violations and the corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at 100 percent rated thermal power (RTP) for the entire reporting period.

Unit 2 began the reporting period shutdown for repairs to the main condenser. The unit was started up on July 23, 2017, but was shutdown to hot standby later that day due to equipment problems. On July 25, 2017, startup resumed, but the reactor was tripped before criticality due to rod position indication problems during the startup. Startup commenced again on July 27, 2017, but was stopped due to additional rod position indication problems. Unit 2 started up after rod position indication repairs on July 30, 2017, and achieved 29 percent RTP on August 2, 2017. The unit remained at that power until August 8, 2017, when the turbine was tripped due to a steam leak on a turbine drain line. The unit stabilized at 8 percent RTP and remained there until power ascension resumed after drain line repairs. Unit 2 reached 100 percent RTP on August 8, 2017, and remained there for the remainder of the reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

External Flood Protection Inspection

a. Inspection Scope

The inspectors reviewed the licensee’s readiness to cope with external flooding. External flooding from a probable maximum precipitation (PMP) or design basis flood (DBF) had the potential for internal flooding of a portion of a number of the plant structures. The inspectors reviewed the feasibility of the licensee’s flooding mitigation plans and design features and verified that they were consistent with the licensee’s design requirements and the risk analysis assumptions for coping with this type of event. The inspectors performed walkdowns of selected areas to observe grading, yard drains, and curbs in the vicinity of the south valve vault rooms. The inspectors also checked status of the flood mode boat. The inspectors reviewed external flood protection features at the intake pumping station and condition of the strainer room sump pumps. Additionally, the inspectors reviewed the licensee’s related corrective action documents (condition reports) to ensure any non-conforming conditions related to potential flooding were properly addressed. The inspection was performed prior to the expected rainfall from Hurricane Irma. This activity constituted one Adverse Weather Protection 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 prior to unit transition into the mode of applicability for the systems. 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 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 six inspection samples, as defined in IP 71111.04.

- 2A and 2B train of motor-driven auxiliary feedwater and Unit 2 turbine-driven auxiliary feedwater prior to mode change
- 2A and 2B train of safety injection prior to mode change
- 2A train of containment spray prior to mode change
- 2B train of containment spray prior to mode change
- 2A-A emergency diesel generator prior to mode change
- 2B-B emergency diesel generator prior to mode change

b. Findings

No findings were identified.

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 three inspection samples, as defined in IP 71111.05AQ.

- Auxiliary building elevation 713’
- Auxiliary building elevation 676’
- Control building elevation 729’ and 741’ (cable spreading room)

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification and Performance (71111.11)

.1 Licensed Operator Requalification Review

a. Inspection Scope

On September 12, 2017, the inspectors observed licensed operator training examinations on the simulator per scenario 3-OT-SRE-1017, revision 7. The scenario included a feedwater line break and subsequent loss of all main and auxiliary feed capability. The inspectors specifically evaluated the following attributes related to the operating crews’ performance:

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of abnormal operating instructions and emergency operating instructions
- Timely and appropriate Emergency Action Level declarations per emergency plan implementing procedures
- 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. 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 used 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.

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 one Maintenance Effectiveness inspection sample, as defined in IP 71111.12.

- Condition Report (CR) 1316520, Unit 2 function 063-B Train A (2A safety injection pump) exceeded performance criteria

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors evaluated, as appropriate, for 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; NPG-SPP-09.11.1, Equipment Out of Service Management; and TI-124, Equipment to Plant Risk Matrix. This activity constituted four Maintenance Risk Assessment inspection samples, as defined in IP 71111.13.

- Risk assessment for August 11, 2017, with the 1A emergency diesel generator (EDG) out of service (OOS) for an extended planned maintenance outage and applicability of TS 3.8.1.B.5 for the extended limiting condition for operation time period based on FLEX EDG availability
- Risk assessment for August 4, 2017, with 1B-B auxiliary feedwater train OOS and replacement main transformer movement under dedicated offsite power lines
- Risk assessment for August 29, 2017, with both sources of offsite power inoperable due to a disqualified grid
- Risk assessment for work week 0905 with 1A-A motor driven auxiliary feedwater, 1A-A component cooling system pump OOS for maintenance and high risk work on Unit 1 turbine electrohydraulic controls, and A main control room chiller OOS

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 significant determination process (SDP). The inspectors verified that the operability evaluations were performed in accordance with NPG-SPP-03.1, CAP. Additional documents reviewed are listed in the Attachment. This activity constituted seven Operability Evaluation inspection samples, as defined in IP 71111.15.

- Immediate determination of operability (IDO) for CR 1320214, momentary indication of Unit 2 reactor rod control bank A rod L5 fully inserted
- Prompt determination of operability (PDO) for CR 1320012, Unit 2 intermittent solid state protection system (SSPS) train B general warning alarm
- Past operability evaluation (POE) for CR 1303309, Unit 1 steam generator 1 and 2 power operated relief valve nitrogen supply found isolated
- PDO for CR 1322853, 2B1 emergency diesel generator engine lube oil circulating pump shaft shear
- PDO for CR 1316395, ERCW system design bases and procedural errors potentially impacting system function
- POE for CR 1316395, ERCW system design bases and procedural errors potentially impacting system function
- Review of CR 1333550, emergency diesel generator 2B inoperable due to low crankcase oil level

b. Findings

.1 Failure to Maintain Procedures for Response to a Loss of Coolant Accident

Introduction. An NRC-identified Green NCV (NCV) was identified for the failure to maintain written procedures as required by TS 5.7.1.1.a. Emergency procedures 1-E-1, revision 7, and 2-E-1 revision 0, both titled Loss of Reactor or Secondary Coolant, contained steps that would have reduced ERCW flow to the A and B CCS HXs and potentially impacted the operability of the A train header of ERCW and CCS for both units.

Description. During an NRC review of a licensee-identified issue regarding the CCS heat exchanger (HX) ERCW outlet and outlet bypass valves, the inspectors found that emergency procedures 1-E-1and 2-E-1 both included a step that directed opening valve 1-FCV-67-458, CCS HX A supply from ERCW header 1B, during a loss of either A train or B train power. This procedural action would be implemented during a loss of coolant accident (LOCA) on one unit with a coincident single active failure causing a loss of train

(A or B) power while the other unit was using the residual heat removal (RHR) system for decay heat cooling. These conditions were incorporated into the design bases for Unit 2 during plant licensing. Procedure 2-E-1 was created with the inappropriate steps on October 8, 2015. Procedure 1-E-1 was updated with identical steps on December 28, 2015. The licensee removed the inappropriate steps in both procedures. The licensee evaluated the past operability of the ERCW system for the time period where the steps were incorporated into the procedure and determined that the condition resulted in the A train of ERCW/CCS being inoperable for Unit 2 for 11 days.

Analysis. The failure to maintain written procedures for emergencies as required by TS 5.7.1.1.a was a performance deficiency. The performance deficiency was more than minor because it affected the Mitigating Systems Cornerstone attribute of Procedure Quality and adversely affected the cornerstone objective in that reduced ERCW flow caused by the inappropriate steps resulted in the Unit 2A train of ERCW/CCS being inoperable for 11 days. This finding was assessed using NRC inspection Manual Chapter 0609, Attachment 4, “Initial Characterization of Findings.” Using Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the finding was determined to require a detailed risk evaluation because it represented an actual loss of function of at least a single train for greater than its TS allowed outage time when the 2A train of ERCW/CCS was inoperable for 11 days. A regional SRA performed the detailed risk evaluation using SAPHIRE Version 8.1.6 and Version 8.50 of the SPAR Model for both units combined. The SRA modified the fault trees for the ERCW 1B & 2A Supply Headers to reflect the inappropriate steps for opening Valve 1-FCV-67-458 given a power loss of either A or B train power, assumed the affected header would fail if the valve were opened, and used an exposure time of one year. The result was less than 1E-6 for each unit which would be a finding of very low significance (Green). For Unit 1, the dominant sequences were related to loss of offsite power where the performance deficiency fails ERCW Header 2A leading to loss of seal cooling. For Unit 2, the dominant sequences were similar with the performance deficiency failing ERCW Header 1B. The risk was mitigated because the performance deficiency would affect operation only when a LOCA occurred with the simultaneous loss of two shutdown boards.

The finding had a cross-cutting aspect in the Documentation attribute of the Human Performance area because the licensee did not maintain the accuracy of 1-E-1 through its revisions and did not maintain procedure 2-E-1 accurate at its creation. (H.7).

Enforcement. TS 5.7.1.1.a, “Procedures,” required, in part, that written procedures be established, implemented, and maintained covering activities related to procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, revision 2, Appendix A, Section 6, “Procedures for Combating Emergencies and Other Significant Events” recommends procedures for loss of coolant. Contrary to the above, since October 8, 2015, 2-E-1, revision 0, was not properly established when a procedural step directing opening of valve 1-FCV-67-458 was included. Also, since December 28, 2015, procedure 1-E-1, revision 7, was not maintained when the same procedural step was added. This violation was entered in to the licensee’s CAP as CR 1331422 and procedures 1-E-1 and 2-E-1 have been revised to remove this step.

This violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy and is identified as NCV 05000390, 391/2017003-01, Failure to Maintain Procedures for Response to a Loss of Coolant Accident.

.2 Inadequate Procedure for Unit Cooldown from Hot Standby to Cold Shutdown

Introduction: An NRC-identified finding of very low safety significance (Green) and associated NCV of TS 5.7.1.1.a, "Procedures," was identified for the failure to maintain TVA procedures 1-GO-6 and 2-GO-6, both entitled Unit Shutdown from Hot Standby to Cold Shutdown. The licensee failed to update the procedures based on a PDO to include steps that would shutdown the running motor driven auxiliary feedwarer pump (MDAFW) prior to starting a third ERCW pump during the period where the opposite unit has been shutdown less than 48 hours.

Discussion: TVA design change notification (DCN) 62151 was issued to ensure the dual unit system alignment and flow settings for the ERCW system would support operability and conform to the design bases for both units as Unit 2 transitioned from construction to full commercial operation. The DCN identified procedural changes necessary to comply with Unit 1 license amendment 104, which added TSs 3.7.16, Component Cooling System – Shutdown, and 3.7.17, Essential Raw Cooling Water System – Shutdown, and the Unit 2 operating license. TS 3.7.16 and 3.7.17 required additional CCS and ERCW pumps to be operable within 48 hours of a unit shutdown. One of the procedure changes discussed in DCN 62151 was necessary to ensure the ERCW system was able to meet the limiting design bases event discussed in Unit 1 license amendment 104 and the Unit 2 operating license which consisted of a design bases LOCA on one unit coincident with a dual unit LOOP, while the other (non-accident) unit is on RHR shutdown cooling within 48 hours after shutdown and experiences a single active failure in the form of a loss of power to one train. The changes consisted of procedure revisions to require starting a third ERCW pump and having provisions to load it as the second ERCW pump on a single diesel generator (EDG) during the limiting design basis event. It was recognized, during the license amendment process, that the diesel generator loading analysis assumed the MDAFW pump was not running on the non-accident unit. However, the limiting design bases event assumes a dual unit LOOP where MDAFW pumps would be automatically loaded onto the non-accident unit's EDGs. As a result, DCN 62151 required the emergency procedures be revised to direct the MDAFW pumps for the non-accident unit be stopped and placed in pull to lock and then activate the applicable ERCW pump interlock bypass switch.

On July 12, 2017, the licensee identified that a previously unknown and unanalyzed failure mode may be more limiting than the limiting design bases event. As part of this discovery the licensee realized the procedural changes in DCN 62151 had not been implemented despite Unit 2 starting commercial operation in September of 2016. As a result, several emergency procedures did not reflect the required ECRW valve position and flow requirements to properly mitigate a limiting design bases accident on Unit 2. The licensee completed a PDO on July 16, 2017. The PDO identified four compensatory actions necessary to restore operability. The four actions were all associated with Unit 1 and Unit 2 emergency and general operating procedure changes.

The inspectors reviewed the PDO and determined that the need to stop a running MDAFW pump prior to loading an EDG with a second ERCW pump, to prevent overloading of the EDG, was not recognized as a required compensatory action to restore operability. The licensee agreed that the procedure changes to stop the running MDAFW pump were required and they revised the PDO on July 17, 2017, to include the necessary procedure changes.

Analysis: The licensee's failure to maintain TVA procedures 1-GO-6, revision 8 and 2-GO-6, revision 6 was a performance deficiency. The performance deficiency was more than minor because it affected the Mitigating Systems Cornerstone attribute of Equipment Performance and affected the cornerstone objective in that failure to maintain the procedures resulted in a condition where the EDG would have been overloaded and rendered inoperable in response to a design basis event. The inspectors evaluated the significance of this finding using IMC 0609, Attachment 4, Appendix A, Exhibit 2, and determined that this finding was of very low safety significance (Green) because the finding did not represent an actual loss of function of a single train for greater than its TS allowed outage time.

The finding had a cross-cutting aspect in the Avoid Complacency component of the Human Performance area as defined in NRC IMC 0310 because the organization failed to recognize the possibility of mistakes and use appropriate error reduction tools. [H.12].

Enforcement: TS 5.7.1.1.a, "Procedures," required, in part, that written procedures be established, implemented, and maintained covering activities related to procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Section 2(j), "General Plant Operating Procedures," required procedures for Hot Standby to Cold Shutdown. Contrary to the above, from July 16, 2017 to July 17, 2017, the licensee failed to maintain their procedures for unit shutdown from hot standby to cold shutdown, 1-GO-6, revision 8 and 2-GO-6, revision 6, because they did not include steps to prevent an EDG overload by stopping the running MDAFW pump. The licensee's immediate corrective actions included revising both procedures to add the required steps. This violation was entered into the CAP as CR 1318176 and is being treated as an NCV, consistent with Section 2.3.2.a of the Enforcement Policy. It is identified as NCV 05000391, 390/2017003-02, Inadequate Procedure for Unit Cooldown from Hot Standby to Cold Shutdown.

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 118921021, 2-SI-68-120, 184 day channel operational test reactor coolant flow loop 3 channel III, loop 2-LPF-68-48D (F-436)
- WO 118851496, 2-SI-99-10-B, 62 day functional test of SSPS train B and reactor trip breaker B following tester circuit board replacement
- WO 118921021, 2-SI-68-120, 184 day channel operational test reactor coolant flow loop 3, channel III, loop 2-LPF-68-48D (F436) following EAGLE 21 DFP circuit board replacement
- WO 119010949, 1-SI-30-902-A, Valve full stroke exercising during plant operation ventilation train A following replacement of quick exhaust valve on 1-FCV-30-40
- WO 118985349, Post maintenance test following 2B2 EDG auxiliary lube oil pump replacement

b. Findings

No findings were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Unit 2 Forced Outage (July 1, 2017 – August 8, 2017)

a. Inspection Scope

The Unit 2 began a forced outage on March 23, 2017, due to a structural failure of the B condenser waterbox. On July 1, 2017, the unit was in mode 5 until the unit began to heat up in preparation for startup. The reactor became critical on July 23, 2017, but returned to hot standby (Mode 3) due to equipment problems with the main feed pumps. On July 25, 2017, startup resumed, but the reactor was tripped before criticality due to rod position indication problems. Startup recommenced on July 27, 2017, but was stopped due to additional rod position indication problems. On July 30, 2017, Unit 2 started up after rod position indication repairs and achieved 29 percent rated thermal power (RTP) on August 2, 2017. The unit remained at 29 percent RTP until August 3, 2017, when the turbine was tripped due to a steam leak on a turbine drain line. The reactor stabilized at 8 percent RTP and remained there until power ascension resumed after drain line repairs. Unit 2 reached 100 percent RTP on August 8, 2017, and remained there for the remainder of the reporting period.

The inspectors observed the licensee's mode changes and startups in order to verify that they were performed in accordance with station procedures and TSs. The inspectors made entry into containment prior to the unit restart to assess the material condition of SSCs, including the containment sump. The inspectors attended forced outage meetings and reviewed the daily risk assessments and condenser repair plans. The inspectors also observed the performance of some surveillance testing being performed while the unit was shutdown. This activity constituted one Refueling and Other Outage Activities sample, 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 ten Surveillance Testing inspection samples; three in-service and seven routine; as defined in IP 71111.22.

In-Service Test:

- WO 118371917, 1-SI-62-901-A, Centrifugal charging pump 1A-A quarterly performance test
- WO 118086192, 2-SI-67-908-B, Valve full stroke exercising and position indication verification during cold shutdown - essential raw cooling water (train 2B)
- WO 118431243, 1-SI-74-901-A, Residual heat removal pump 1A quarterly performance test

Other Surveillances

- WO 118431170, 0-SI-82-12-A, Monthly diesel generator start and load test DG 2A-A
- WO 118086055, 2-SI-0-710, Containment integrity: penetrations
- WO 117823693, 2-SI-211-1-A, 18 month 6.9 KV shutdown board 2A-A automatic and manual transfer tests
- WO 118061393, 2-SI-211-1-B, 18 month 6.9 KV shutdown board 2B-B Automatic and Manual Transfer Tests
- WO 117823686, 2-SI-211-3-A, 18 month functional test on 6900V SD BD 2A-A degraded and undervoltage relays
- WO 117823687, 2-SI-211-3-B, 18 month functional test on 6900V SD BD 2B-B degraded and undervoltage relays
- WO 117823601, 2-SI-68-86, 18 month channel calibration of remote shutdown monitoring narrow range pressurizer pressure loop 2-LPP-68-337C

b. Findings

Introduction: A self-revealed finding of very low safety significance (Green) and associated NCV of TS (TS) 5.7.1.1.a, “Procedures,” was identified for the failure to follow TVA procedure 2-SI-68-86, 18 Month Channel Calibration of Remote Shutdown Monitoring Narrow Range Pressurizer Pressure Loop 2-LPP-68-337C, Revision 4. The licensee failed to properly follow step 6.2.6 [1.3], which resulted in the inadvertent lifting of a pressurizer power operated relief valve (PORV).

Discussion: On June 21, 2017, instrumentation and control technicians were performing Surveillance 2-SI-68-86. The surveillance verified the function of the transfer switches for the PORV and its associated block valve to transfer power from the main control room to the auxiliary control room. Step 6.2.6 [1.3] of the procedure directed that the distributed control system (DCS) demand for the PORV be toggled to 0 (closed). When the technicians came to this step, they toggled the output as directed in the beginning of the procedure step. However, they did not recognize that the DCS demand was at 0 and, therefore, toggled it to 1 (open). When the auxiliary transfer switch was operated, the PORV had an open signal present and opened. This resulted in a reactor coolant pressure drop from 335 psig to 310 psig. The main control room operators were alerted to this condition by an annunciator for high pressure in the pressurizer relief tank, properly diagnosed the inadvertent PORV opening, and shut the associated PORV block valve stopping the pressure decrease.

Analysis: The licensee’s failure to follow TVA procedure 2-SI-68-86, was a performance deficiency. The performance deficiency was more than minor because it affected the Initiating Events Cornerstone attribute of Human Performance and adversely affected the cornerstone objective in that failing to follow procedure 2-SI-68-86 resulted in a temporary lowering of reactor coolant pressure and inventory. The finding was screened in accordance with NRC IMC 0609, Attachment 4, Appendix G, “Shutdown Operations Significance determination process Phase 1 Initial Screening and Characterization of Findings.” The finding was screened to Green based on the answers to questions 2 and 3. The resultant leakage from the open PORV would not have caused the current decay heat removal method to fail if it went undetected and leakage would be self-limiting such that it would stop before impacting the operating method of decay heat removal.

The finding had a cross-cutting aspect in the Challenge the Unknown component of the Human Performance area as defined in NRC IMC 0310, because the technicians failed to recognize that the output was already set to 0, but proceeded anyways to toggle the output which resulted in setting it to 1 [H.11].

Enforcement: TS 5.7.1.1.a, “Procedures,” required, in part, that written procedures be established, implemented, and maintained covering activities related to procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Section 8, “Procedures for Control of Measuring and Test Equipment and for Surveillance Tests, Procedures, and Calibrations” requires procedures for surveillance tests. Contrary to the above, required surveillance procedure 2-SI-68-86, revision 4, was not implemented when step 6.2.6 [1.3] was not performed as written. Corrective actions taken or planned by the licensee include revisions to 2-SI-68-86 to clarify the steps relating to toggling the DCS output, training for the craft, and management oversight of pre-job briefs. This violation was entered into the CAP as CR 1309345 and is being treated as an NCV, consistent with Section 2.3.2.a of the Enforcement Policy. This violation is identified as NCV 05000391/2017003-03, Failure to Follow a Surveillance Procedure Led to an Inadvertent Lift of a Pressurizer Power Operated Relief Valve.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

On the dates listed below, the inspectors observed a licensee-evaluated emergency preparedness drill to verify that the emergency response organization was properly classifying the event in accordance with licensee procedure EPIP-1, Emergency Plan Classification Flowchart, and making accurate and timely notifications and protective action recommendations in accordance with EPIP-2, Notification of Unusual Event; EPIP-3, Alert; EIPIP-4, Site Area Emergency; EPIP-5, General Emergency; and the Radiological Emergency Plan. In addition, the inspectors verified that licensee evaluators were identifying deficiencies and properly dispositioning performance against the performance indicator criteria in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline. This activity constituted two EP drill evaluation inspection samples.

- EP drill on July 17, 2017
- EP drill on August 16, 2017

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 two 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 two performance indicator samples, as defined in IP 71151.

- High Pressure Safety Injection MSPI
- RCS leak rate

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Review of Items Entered into the CAP

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 CAP. This review was accomplished by reviewing daily condition report (CR) summary reports and attending daily CR review meetings

.2 Annual Sample: Review of CR 129727, Watts Bar Elevation Letter – Operations Leadership Formality and Rigor

a. Inspection Scope

The inspectors reviewed CR 1297271, WBN Elevation Letter – Operations Leadership Formality and Rigor, in detail to evaluate the effectiveness of the licensee’s corrective actions intended to address operator performance concerns. The CR was written to address the continued lack of formality, rigor, and discipline by operators in monitoring and controlling the plant. The inspectors assessed whether issues were properly identified, documented accurately and completely, properly classified and prioritized, adequately considered extent of condition, generic implications, common cause, and previous occurrences, adequately identified root causes/apparent causes, and identified appropriate and timely corrective actions. The inspector reviewed processes contained in the licensee’s Conduct of Operations procedure (OPDP-1) and CAP (NPG-SPP-22.300). This activity constituted one sample of in-depth review as defined in IP 71152.

b. Observations and Findings

To address the concerns identified in CR 1297217, the licensee developed a High Intensity Training (HIT) program. The training was developed to refocus training personnel and license operators of standards, behaviors and expectations associated with plant operations. The inspector discussed the licensee’s HIT program with members of the licensee’s training staff, operation’s management, and licensee operators during a four day period. During the discussions, the inspector was able to obtain a clear understanding of why and how HIT was developed.

During the four days of observing HIT activities, the inspectors observed two operating crews and two crews of evaluators in a training environment. The inspector also observed classroom training and critiques following each simulator scenario. Many of the training activities were also observed by a member of the licensee's corporate training staff, onsite operations management, a contract third party evaluator, and a peer evaluator from another utility.

The training sessions were found to be very intense and operational focused. The evaluators were extremely critical of crew performance. The evaluators took every opportunity to identify and address concerns. Whenever a concern/issue was identified, the scenario was stopped and the issues was discussed with the crew. Stopping the scenario and holding discussions occurred numerous times throughout each scenario. Following each discussion, the simulator was reset to the desired point and reran. The discussions were very interactive. During the discussions, the evaluators constantly focused on procedural requirement and licensee expectations. The evaluators were often challenged/questioned by crew members. The evaluators adequately addressed each question or concern identified by the crew. The inspector also observed critiques following scenarios.

From the inspector's observation it was clear that HIT was designed to address operational performance issues identified in the CR. The effectiveness of HIT can only be evaluated by observing operator and plant performance over time. The inspectors concluded that the training provided during HIT, if embraced, should decrease lack of formality, increase rigor, and improve discipline by operators in monitoring and controlling the plant. The HIT would also be expected to improve operators' implementation of standards outlined in OPDP-1, Conduct of Operations. The inspectors will continue to monitor operator and plant performance in the control room, during actual plant events and in licensed operator simulator training, as required by the baseline inspection program. No findings were identified.

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The review was focused on trends in risk management, long-standing minor equipment deficiencies, housekeeping, TS compliance, corrective action screening and condition adverse to quality documentation.

b. Observations and Findings

No findings were identified. The inspectors had several observations regarding the trends listed above. Regarding risk management, the inspectors noted that the environmental factor for the equipment out of service computer program (EOOS) was not consistently adjusted per procedure to reflect activities in the plant switchyard. This was initially identified to the licensee in 2016. The condition report written at that time documented the issue as an NRC question, rather than a failure to follow the EOOS procedure, and the corrective action was to respond to the NRC to ensure that their question was answered, rather than address procedure non-compliance. The inspectors re-visited this with the licensee when they observed switchyard work in progress without

the environmental factor setting in EOOS being per procedure. This time the licensee properly characterized the issue as procedure non-compliance in their CAP. The inspectors used the EOOS test module and verified that risk remained GREEN during instances when the environmental factor adjustment was not properly set. The inspectors noted that, for the work performed when the environmental factor was not properly set, the licensee did implement physical risk mitigation controls at the work sites that were in accordance with the appropriate work management procedures.
The inspectors also noted a trend in long-standing equipment issues eventually becoming either operator distractions or worse conditions. In one instance valve leakby in the chemical volume and control system gave erroneous indication that the reactor coolant system was either being borated or diluted. This required the operating crew to enter procedures to then verify that the RCS truly was neither borated nor diluted. In another instance, known leakage on the 1A high pressure fire pump shaft seal worsened to the point that protective measures had to be taken to shield water spray from the power supply conduit of the pump.

Since the completion of Unit 2 construction, the inspectors noted a reduction in the amount of temporary equipment stored in the plant auxiliary building and general housekeeping improvements in the auxiliary building. CAP review during the first and second quarter of 2017 showed a more aggressive approach by the license in improving housekeeping and removing lingering temporary equipment. Documents reviewed show that the licensee accomplished this through frequent health and safety walkdowns and challenging temporary equipment tags that were out of date. The inspectors observed the results of these efforts in their routine walkdowns of risk-significant areas. Specifically, in regards to a large scaffold storage area near the Unit 2 713 level penetration. Although temporary equipment tags were present and up to date, the area appeared to have become a convenient location to temporarily store a wide variety of items beyond scaffolding. The licensee identified this in their CAP and then completely removed all of the items stored in the area.

The inspectors also identified negative trends in the treatment of C-level CRs in the CAP and with TS compliance issues. Inspectors identified multiple C-level CRs during the inspection period that exhibited one of the following issues: inadequate documented condition details; inadequate screening of conditions adverse to quality (CAQs) to non-CAQ status; and failure to promptly identify CAQs. Inspectors also noted several examples of issues with TS compliance and proper TS application during the inspection period. The licensee has identified these issues in their CAP.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report (LER) 05000390, 391/2016-010-00, Emergency Diesel Generator Crankcase Pressure Switches Not Analyzed to Withstand the Effects of a Tornado

A condition involving the potential impact of a tornado on the EDGs was identified during an NRC Component Design Basis Inspection at the Sequoyah Nuclear Plant. The EDGs were designed with a crankcase pressure trip setpoint of approximately one inch of water which is bypassed during an emergency start. A tornado could potentially induce a pressure spike which could cause actuation of the crankcase pressure trip due to different vent paths between the EDG room and the EDG crankcase. Actuation of the crankcase pressure trip would energize the shutdown relay causing an EDG lockout condition. The EDG lockout condition would prevent all EDG starts until operators manually reset the lockout condition. Because the EDGs at Watts Bar were essentially identical designs, this condition was reviewed for applicability to Watts Bar. The licensee determined this condition placed both units in an unanalyzed condition that could have potentially affected all four EDGs simultaneously. This was a legacy EDG protective logic circuitry design that did not anticipate the interaction between the crankcase pressure trip and the outside atmospheric pressure spike during a tornado. This condition was documented in the licensee CAP as CR 1179264. A compensatory action was established of starting the EDGs in the emergency mode when notified of a Tornado Warning and ran while the Tornado Warning was in effect ensuring the EDGs would be available to perform their required safety function. The licensee also implemented DCN 66376 to remove the seal-in function of the crankcase differential pressure switches and retain the alarm function of the switches for all four EDGs. This LER was reviewed by the inspectors. A licensee-identified violation is documented in Section 4OA7.

.2 (Closed) LER 05000390/2016-001-00, Channel Mode Switch in Incorrect Position Renders Lower Containment Atmosphere Particulate Radiation Monitor Inoperable.

a. Inspection Scope

On January 12, 2016, at 1645 Eastern Standard Time (EST), Watts Bar Nuclear Plant (WBN) Maintenance personnel were performing a 92 day channel operational test for radiation monitor 1-RM-90-1064, Lower Containment Atmosphere Particulate Radiation Monitor, and found the mode switch in the "DIFF" position, which was not expected. The surveillance was stopped and an investigation was conducted. It was determined that the design required the mode switch to be in the "INT" position to be operable. The mode selector switch was placed in the "INT" position and the surveillance was completed. The radiation monitor was restored to OPERABLE status at 1743 EST on January 12, 2016. Placing the mode selector switch in the "DIFF" position resulted in 1-RM-90-1064 being INOPERABLE due to the loss of alarm function of the monitor. Investigation determined that the switch had been repositioned on December 8, 2015. Because the containment particulate radiation monitor was inoperable for a period of time greater than permitted by TS 3.4.15, this condition was reportable as an operation or condition prohibited by TS per 10 CFR 50.73(a)(2)(i)(B). During the time the monitor was inoperable, other means of leak detection (e.g., containment pocket sump level indication, reactor coolant system inventory balance) remained available. This LER was reviewed by the inspectors. No additional findings or violations of NRC requirements were identified.

.3 (Closed) LER 05000390/2016-005-00, Both Trains of Unit 1 Emergency Gas Treatment System Inoperable During Unit 2 Testing

On March 14, 2016, Watts Bar Nuclear Plant (WBN) Unit 1 determined through engineering analysis that both trains of emergency gas treatment system (EGTS) were inoperable for 8 minutes, 10 seconds during preoperational testing of Unit 2 EGTS. The inoperability of A and B trains of Unit 1 EGTS took place on October 22, 2015, while Unit 1 was in Mode 1 and two trains of EGTS were required to be operable in accordance with TS LCO 3.6.9, "Emergency Gas Treatment System (EGTS). At the time of the event, Unit 2 was in "no mode," prior to initial fuel loading. With both trains of EGTS inoperable, the specified safety functions of Unit 1 EGTS were not capable of being performed. Therefore, this condition was reported pursuant to 10 CFR 50.73(a)(2)(v)(C) and 10 CFR 50.73(a)(2)(v)(D), "Event or Condition That Could Have Prevented Fulfilment of a Safety Function." This LER was reviewed by the inspectors. No additional findings or violations of NRC requirements were identified.

.4 (Closed) LER 05000390/2016-004-00, Automatic Reactor Trip Due to Actuation of Over Temperature Delta Temperature Bistables

On March 22, 2016, at 1131, Watts Bar Nuclear Plant Unit 1 experienced an automatic reactor trip. The initiating reactor trip first out received was 76-C Over-temperature Delta T. The turbine trip first out received was 73-C Rx Trip Breakers RTA and BYA Open. Prior to the unit trip, Unit 1 was in Mode 1 at 100 percent power. Concurrent with the reactor trip, the auxiliary feedwater system actuated. All control rods inserted upon the reactor trip and safety systems functioned as expected. This LER was reviewed by the inspectors. No additional findings or violations of NRC requirements were identified.

.5 (Closed) LER 05000390/2016-006-00, Undersized Room Cooler Fan Shaft Results in Loss of Centrifugal Charging Pump

On May 13, 2016, Watts Bar Unit 1 determined that a condition prohibited by TSs had previously occurred. During the Fall 2015 outage, maintenance performed on the 1B-B centrifugal charging pump (CCP) room cooling fan introduced a condition that resulted in a subsequent bearing failure of the room cooling fan. This condition would have prevented the 1B-B CCP pump from performing its function for its designed mission time. Based on the reduced reliability of the fan, the 1B-B CCP was considered to be inoperable from October 7, 2015, until the fan was repaired and returned to service on December 6, 2015. During this time, there were several short periods when the 1A-A CCP was also inoperable. A NCV for this condition was documented in NRC Inspection Report 05000390, 391/2016002-02. The LER was reviewed by the inspectors. No additional findings or violations of NRC requirements were identified.

.6 (Closed) LER 05000390/2016-011-00, Loss of Centrifugal Charging Pump Due to Repeat Failure of Associated Room Cooler

On August 3, 2016, Wafts Bar Nuclear Plant Unit 1 (WBN1) determined that a condition prohibited by TS had previously occurred. During maintenance of the 1B-B CCP room cooler, the bearing was found in a degraded condition requiring repair. This fan was required to support Operability of the 1B-B CCP. The fan had been previously repaired on December 6, 2015, and had less than 100 days of operation since its overhaul. The mission time of the CCPs is specified in design documents as 100 days. Based on the inability of the CCP to meet its mission time, the 1B-B CCP was considered to be design inoperable since its overhaul on December 6, 2015. This represents a condition prohibited by TS for the 1B-B CCP being inoperable for greater than its allowed outage time. The LER was reviewed by the inspectors. No findings or violations of NRC requirements were identified.

4OA5
.1 IP 93100 Safety-Conscious Work Environment Issue of Concern Follow Up

a. Inspection Scope

The inspectors assessed the TVA Nuclear corporate safety-conscious work environment (SCWE) by conducting safety culture interviews of individuals from the engineering, licensing, and operations groups. Inspectors interviewed a total of 22 individuals to determine if indications of a chilled work environment exist, employees are reluctant to raise safety and regulatory issues, and employees are being discouraged from raising safety or regulatory issues. Information gathered during the interviews was used in aggregate to assess the work environment at TVA Nuclear corporate.

b. Assessment

Based on the interviews conducted, the inspectors determined that licensee management emphasized the need for all employees to identify and report problems using the appropriate methods established within the administrative programs, including the CAP and Employee Concerns Program. These methods were readily accessible to all employees. Based on discussions conducted with a sample of employees from various departments, the inspectors determined that employees felt free to raise safety and regulatory issues, and that management encouraged employees to place issues into the CAP for resolution. The inspectors did not identify any reluctance on the part of the licensee staff to report safety concerns.

4OA6 Meetings, including Exit

On October 25, 2017 and November 8, 2017, the resident inspectors presented the inspection results to 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 licensee-identified violations of NRC requirements were determined to be of very low safety significance and met the NRC Enforcement Policy criteria for being dispositioned as NCVs.

- Technical Specification 5.7.1.1.a, “Procedures,” required, in part, that written procedures be established, implemented, and maintained covering activities related to procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Revision 2, Appendix A, Section 6, "Procedures for Combating Emergencies and Other Significant Events" requires procedures for a reactor trip. Contrary to the above, from May 23, 2016, until July 16, 2017, procedure 2-E-0, Revision 5, Reactor Trip and Safety Injection, was not maintained which resulted in a condition where CCS Heat Exchanger B (ERCW/CCS Train 2A) would not have been able to remove sufficient heat during sump recirculation following a LOCA on Unit 2 for approximately 75 days. This condition was caused by the licensee's failure to implement ERCW system DCN 62151 as written. A detailed risk evaluation was performed using SAPHIRE Version 8.1.5 and Version 8.50 of the SPAR Model for both units combined. The result was less that 1E-6/year for Unit 2, which would be a finding of very low significance (Green). This violation was entered in to the licensee's CAP as CR 1316395.

- Technical Specification 5.7.1.1.a stated, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures in Regulatory Guide 1.33 Rev. 2, Appendix A, February 1978. Procedures for locking and tagging are applicable procedures under REG GUIDE 1.33 Appendix A, 1.c Equipment Control. Contrary to this requirement, Step 3.2.4.M of procedure NPG-SPP-10.2, Clearance Procedure to Safely Control Energy, Revision 18 was not followed when nitrogen supply isolation valves 2-ISIV-1-408L and 2-ISIV-1-408M and isolation valves 2-ISIV-1-405L and 2-ISIV-1-405M were closed and tagged but not documented as tagged in the Electronic Shift Operations Management System (eSOMS). As a result, the valves remained closed resulting in the inability to operate the Unit 2 SG#1 and #2 PORVs using back-up nitrogen. The finding was determined to be Green because having the nitrogen supply to two out of four steam generator PORVs isolated only affects the ability to achieve and maintain cold shutdown. The licensee documented this violation as CR 1303309.

- Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," required, in part, a testing program to demonstrate that quality related SSCs will perform satisfactorily in service and performed in accordance with written test procedures. Contrary to the above, from at least 2010 until July 2017, various safety-related valves were unacceptably preconditioned prior to required as-found testing. This finding was of very low safety significance (Green) because the finding did not represent an actual loss of function of a single train for greater than its TS allowed outage time. The licensee documented this violation as CRs 1276605, 1316712, 1319298, 1319304.

- 10 CFR Part 50, Appendix B, Criterion III, "Design Control," stated, in part, that, measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of SSCs. Contrary to the above, for at least the past twenty years, the licensee failed to assess the effects of a tornado on the crankcase over-pressure trip which could prevent EDGs from fulfilling their safety-related function. A regional senior reactor analyst performed a detailed risk evaluation and determined the dominant accident sequences involved a weather-related loss of offsite power with all four EDGs failing due to the performance deficiency and the operators recovering one of the failed EDGs. The risk of this performance deficiency was not greater than Green due to the low frequency of tornados/high winds and the potential for operator recovery. The licensee documented this violation as CR 117926.

- Technical Specification LCO 3.6.3, Containment Isolation Valves, required that each containment isolation valve shall be operable in modes 1, 2, 3, and 4. TS Required Action statement 'A.1' required that the affected penetration flow path be isolated, and Required Action 'A.2', directed that the penetration flow path is verified to be isolated once per 31 days. Contrary to the above, on May 18, 2017, containment isolation valve 1-FCV-31-330 was tagged closed for maintenance; however no verification that the flow path was isolated was performed until August 23, 2017. This finding was of very low safety-significance (Green) because it did not represent an actual open pathway in the physical integrity of reactor containment and was not related to hydrogen ignitors. The licensee documented this violation as CR 1331287.

- Unit 1 Operating License condition 2.F required, in part, that TVA shall implement and maintain in effect all provisions of the approved Fire Protection Program. The Fire Protection Report was developed to ensure compliance with the requirements of this licensee condition. Fire Protection Report, Part II, is the Fire Protection Plan (FPP). FPP Subsection 14.10, Fire Safe Shutdown Equipment, required nonfunctional equipment listed in Table 14.10 be restored to its functional status within 30 days. If this 30 day requirement cannot be met, then the equipment be placed in its fire safe shutdown (FSSD) position. Contrary to the above, during a surveillance on June 10, 2017, backdraft damper 0-BKD-31-592, equipment listed in Table 14.10, was identified as not being able to achieve its FSSD position. However, actions to place the damper in its FSSD position were not taken until July 11, 2017. This finding was of very low safety significance because there was a fully functional automatic suppression system on either side of the fire barrier. This violation was documented as CR 1316058.

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

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

LIST OF REPORT ITEMS

<u>Opened and Closed</u>

NCV 05000390, 391/2017003-01	Failure to Maintain Procedures for Response to a Loss of Coolant Accident (Section 1R15.1)
NCV 05000391, 390/2017003-02	Inadequate Procedure for Unit Cooldown from Hot Standby to Cold Shutdown (Section 1R15.2)
NCV 05000391/2017003-03	Failure to Follow a Surveillance Procedure Led to an Inadvertent Lift of a Pressurizer Power Operated Relief Valve (Section 1R22)

<u>Closed</u>

LER 05000390, 391/2016-010-00	Emergency Diesel Generator Crankcase Pressure Switches Not Analyzed to Withstand the Effects of a Tornado (Section 4OA3.1)
LER 05000390/2016-001-00	Channel Mode Switch in Incorrect Position Renders Lower Containment Atmosphere Particulate Radiation Monitor Inoperable (Section 4OA3.2)
LER 05000390/2016-005-00	Both Trains of Unit 1 Emergency Gas Treatment System inoperable During Unit 2 Testing (Section 4OA3.3)
LER 05000390/2016-004-00	Automatic Reactor Trip Due to Actuation of Over Temperature Delta Temperature Bistables (Section 4OA3.4)
LER 05000390/2016-006-00	Undersized Room Cooler Fan Shaft Results in Loss of Centrifugal Charging Pump (Section 4OA3.5)
LER 05000390/2016-011-00	Loss of Centrifugal Charging Pump Due to Repeat Failure of Associated Room Cooler (Section 4OA3.6)

LIST OF DOCUMENTS REVIEWED

<u>Section 1R01: Adverse Weather Protection</u>
0-MI-17.003, Flood Mode Preparation Storage Locations and Periodic Inventory, Rev. 0012
0-TI-444, External Flood Protection Program, Rev. 0003

<u>Section 1R04: Equipment Alignment</u>
<u>Procedures</u>
2-SI-63-8, ECCS Valve Alignment Verification, Rev. 0002
2-SI-3-130, AFW Valve Alignment Verification, Rev. 0004
2-SOI-63.01 ATT 1V, Safety Injection System, Rev. 0005
2-SI-70-1, Component Cooling System, Safety-Related Valves: Alignment Verification, Rev. 0004
2-SOI-72.01, Containment Spray System, Rev. 0005
2-SOI-72.01 Containment Spray System Valve Checklist 2-71.01V, ATT 1V, Rev. 0001
0-SOI-82.03, Diesel Generator (DG) 2A-A, Rev. 0012
0-SOI-82.03, Diesel Generator 2A-A Power Checklist 82.03-1P, ATT 1P, Rev. 0000
0-SOI-82.03, Diesel Generator 2A-A Valve Checklist 82.03-1V, ATT 1V, Rev. 0010
0-SOI-67.01, Essential Raw Cooling Water System Supply Header 1A Valve Alignment Checklist 0-67.01-3V, ATT 3V, Rev. 0017
0-PI-OPS-17.0, 18 Month Locked Valve Verification, Rev. 0082
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Drawing 47W866-11
Drawing 47W920-2
Drawing 47A381-20
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2-SI-68-114, 184 Day Channel Operational Test RCS Flow Loop 1 Channel III Loop 2-LPF-68-6D (F-416), Rev. 0003
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2-SI-67-908-B, Valve Full Stroke Exercising and Position Indication Verification During Cold SD – ERCW (Train 2B), Rev. 0003
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1EP6: EP Drill Evaluation

Controller's package for July 17, 2017, training drill dated 7/17/17
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Section 4OA3: Followup of Events and Notices of Enforcement Discretion

Documentation of Information Sharing – Title: Radiation Meter 1-RM-90-106A
Design Change Notice #66212, Rev. A for Equipment: Various/System 65 (Emergency Gas Treatment System) to revise SDD N3-65-4001 to Incorporate Test Requirements, dated: 2/11/2016
CR 11430756 Level 2 Evaluation Action 007 dated: 07/15/2016
Past Operability Evaluation Documentation for CR 1143076 signed on 3/10/2016.
Routine WO 117688915, Equipment Description: EH Fluid Display Subpanel, Unit 1 Reactor Trip. Dated: 3/22/2016.
Level 2 Evaluation – CR Number 1152462, Rev 0 dated 4/26/2016.
NPG Technical Pre-Job Briefing Checklist AEC CR1152462 dated 3/31/2016
TVA Corrective Action 1152462-006 Completed 12/21/2016.
TVA Condition Report 1152462 draft: 03/22/2016 Unit 1 Reactor Trip
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