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

August 4, 2015

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

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000390/2015002

Dear Mr. Shea:

On June 30, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Watts Bar Nuclear Plant, Unit 1. On July 30, 2015, the NRC inspectors discussed the results of this inspection with Mr. M. Taggart and other members of the Watts Bar staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented three NRC-identified and one self revealing findings of very low safety significance (Green) that involved violations of NRC requirements. 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.

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding,” of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure:
NRC Inspection Report 05000390/2015002
w/Attachment: Supplemental Information

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Letter to Joseph Shea from A. Blamey dated August 4, 2015

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000390/2015002

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REGION II

Docket No.:	50-390
License No.:	NPF-90
Report No.:	05000390/2015002
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Unit 1
Location:	Spring City, TN 37381
Dates:	April 1 through June 30, 2015
Inspectors:	J. Nadel, Senior Resident Inspector J. Hamman, Resident Inspector C. Cheung, Resident Inspector T. Palmer, Senior Reactor Technical Instructor W. Russell, Senior Reactor Technical Instructor R. Hickcok, Senior Reactor Technical Instructor R. Egli, Branch Chief, Reactor Technology Branch A P. Cooper, Reactor Inspector J. Rivera-Ortiz, Senior Reactor Inspector J. Eargle, Senior Reactor Inspector
Approved by:	Alan Blamey, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY

IR 05000390/2015-002; April 1, 2015 – June 30, 2015; Watts Bar, Unit 1; Operability Evaluations, Plant Modifications, Surveillance Testing.

The report covered a three-month period of inspection by the resident inspectors and regional inspectors. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process" Revision 5.

Four Green non-cited violations were identified. The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP) dated April 29, 2015. The cross-cutting aspects are determined using IMC 0310, "Aspects within the Cross-Cutting Areas" dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated February 4, 2015.

<u>Findings and Violations</u>

<u>Cornerstone: Mitigating Systems</u>

- <u>Green</u>: The NRC identified a NCV of 10 CFR 50.55a, "Codes and Standards," for the licensee's failure to meet the test requirements set forth in the American Society of Mechanical Engineers ASME Operation and Maintenance (OM) for Residual Heat Removal (RHR) flow control valves (FCVs). Specifically, TVA failed to scope the RHR FCVs into their In-Service Testing (IST) program. Immediate corrective actions included modifying the RHR pump testing procedures to perform the required remote position indication testing. The licensee entered this issue into their corrective action program as PER 995791.

  The performance deficiency was determined to be more than minor because if left uncorrected, the failure to perform required IST testing could lead to a more significant safety concern in that valve degradation could go unnoticed resulting in undetected inoperability. The inspectors 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 performance deficiency had a cross-cutting aspect of Conservative Bias in the area of Human Performance because the licensee failed to use conservative decision making practices in their evaluation of the status of the RHR FCVs after being challenged by the NRC. (H.14) (1R15.1)

- <u>Green:</u> The NRC identified a NCV of technical specification (TS) 5.7.1.1.a, Procedures, for the licensee's failure to implement OPDP-8, Operability Determinations and Limiting Conditions for Operations (LCO) tracking. Specifically, the licensee failed to track the applicability of action statement 'A' of TS LCO 3.5.2.A, emergency core cooling systems, during planned testing. The licensee entered this issue into their corrective action program as CR 1010269.

The performance deficiency was more than minor because, if left uncorrected, it would have had the potential to lead to a more significant safety concern in that, the failure to track an applicable technical specification action statement could lead to plant operations outside of TS analyzed conditions. The inspectors 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 and did not represent an actual loss of function of one or more non-technical specification equipment for greater than 24 hours. The performance deficiency had a cross-cutting aspect of Avoid Complacency in the area of Human Performance because licensee personnel were complacent and failed to question long held assumptions about the ability of the valves to fail to their safe position under all design basis conditions. (H.12) (1R15.2)

- Green: The NRC identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to consider the effects of a break in the non-seismic portion of the essential raw cooling water (ERCW) discharge flow path to the cooling tower basin in the calculation used to determine the net positive suction head available to the auxiliary feedwater pumps. The licensee entered the issue into their corrective action program as problem evaluation report 97923 and has planned corrective actions to seismically qualify portions of the ERCW discharge path to the cooling towers.

  The performance deficiency was determined to be more than minor because it was associated with the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences by potentially reducing the Net Posittive Suction Head (NPSH) available to the Auxillary Feedwater (AFW) pumps. The inspectors determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component (SSC), and the SSC maintained its operability. The inspectors determined that no cross-cutting aspect was applicable because the finding was not indicative of current licensee performance. (1R21)

- Green. A self-revealing, Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" was identified for the licensee's failure to follow procedure 1-SI-99-10-A, 62 Day Functional Test of Solid State Protection System (SSPS) Train A and Reactor Trip Breaker A, Revision 59 as amended, for troubleshooting by Procedure Control Form 070-4. Specifically, the licensee attempted to take voltage measurements which were not directed by the revised procedure. The licensee stopped testing, conducted a prompt investigation and removed the first line supervisor and foreman from their duties pending remediation. The licensee placed the issue into their corrective action program as CR 1015778

  The performance was more than minor because it adversely affected the equipment performance attribute of the mitigating systems cornerstone to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the failure to follow the troubleshooting procedure resulted in drawing an arc in the SSPS cabinet and tripping an upstream supply breaker which

resulted in the inoperability of the 1A-A containment spray pump. The inspectors 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 of containment spray for greater than its Tech Spec allowed outage time. The performance deficiency had a cross-cutting aspect of Procedure Adherence in the area of Human Performance because crew members failed to follow the work instructions in the troubleshooting procedure (H8). (1R22)

REPORT DETAILS

Summary of Plant Status

The unit started the reporting period at 100 percent rated thermal power and remained there through the end of the reporting period.

1. REACTOR SAFETY
   Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

Review of Offsite and Alternate AC Power System Readiness

a. Inspection Scope

Prior to the summer season, inspectors verified plant features, interviewed control room personnel, and reviewed procedures for operation and continued availability of offsite and alternate AC power systems. Inspectors reviewed the licensee's procedures and interface agreements affecting these areas and the communication protocols between the northeast area dispatcher and the control room to verify that the appropriate information is exchanged when issues arise that could impact the offsite power system and the alternate AC power system. Documents reviewed are listed in the Attachment. This activity constituted one Adverse Weather Protection inspection sample.

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 (OOS). 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. Documents reviewed are listed in the Attachment. This activity constituted five Equipment Alignment Partial System Walkdown inspection samples.

- 1A-A centrifugal charging pump (CCP) while 1B CCP was OOS for maintenance
- Electric board room (EBR) water chiller A-A
- EBR water chiller B-B
- 1B-B ERCW strainer while the 1A-A strainer was OOS for maintenance
- 2B-B ERCW strainer while the 2A-A strainer was OOS for maintenance

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

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. Documents reviewed are listed in the Attachment. This activity constituted 15 Fire Protection Walkdown inspection samples.

- Emergency diesel generator (EDG) 1A-A
- EDG 1B-B
- EDG 2A-A
- EDG 2B-B
- B Train essential raw cooling water (ERCW) pump room
- A Train ERCW pump room
- A Train high pressure fire protection (HPFP) room
- B Train HPFP room
- A Train ERCW strainer room
- B Train ERCW strainer room
- 6.9kV shutdown board room
- Vital DC board room I, II, III, IV (counts as four samples)

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

.1 Annual Review of Heat Sink Performance

a. Inspection Scope

The inspectors performed two heat sink performance reviews. The inspectors reviewed the licensee's program for maintenance and testing of the EBR water chillers. Specifically, the review included the testing and analysis of the EBR water chiller A-A (0-CHR-031-0128) and oil cooler (0-CLR-031-0128); and the EBR water chiller B-B (0-CHR-031-0129) and oil cooler (0-CLR-031-0129). The inspectors reviewed the ERCW system description, the eddy current testing program document, as well as completed work orders (WOs) documenting the testing, visual inspection, online flushing, and associated corrective actions to verify that corrosion or fouling did not impact the heat exchanger from achieving its design basis heat removal capacity. Documents reviewed are listed in the Attachment. This inspection constituted two Heat Sink Performance inspection samples.

b. Findings

No findings were identified.

.2 Triennial Review of Heat Sink Performance (71111.07T)

a. Inspection Scope

The inspectors reviewed, where applicable, vendor manual information, associated calculations, performance test results, and inspection results for the Emergency Diesel Generator (EDG) 1A1-1 Heat Exchanger (HX), DG 1A1-2 HX, Containment Spray (CS) 1A HX, CS 1B HX, Main Control Room A/C Chiller B-B, and the RHR Pump Room Cooler 1A-A. These heat exchangers were chosen based on their risk significance in the licensee's probabilistic safety analysis, and their important safety-related mitigating system support functions.

The inspectors determined, where applicable, whether the testing, inspection, maintenance, and monitoring of biotic fouling and macrofouling programs for the selected heat exchangers were adequate to ensure proper heat transfer. This was accomplished by determining whether the test method used was consistent with accepted industry practices, or equivalent, the test conditions were consistent with the selected methodology, the test acceptance criteria were consistent with the design basis values, and reviewing results of heat exchanger performance testing. The inspectors also determined whether the test results appropriately considered differences between testing conditions and design conditions, the frequency of testing based on trending of test results was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values, and test results considered test instrument inaccuracies and differences.

For the heat exchangers selected, the inspectors reviewed the methods and results of heat exchanger performance inspections. The inspectors determined whether the methods used to inspect and clean heat exchangers were consistent with as-found conditions identified, expected degradation trends, and industry standards. The inspectors also verified the licensee's inspection and cleaning activities had established acceptance criteria consistent with industry standards, and the as-found results were recorded, evaluated, and appropriately dispositioned so that the as-left condition was acceptable.

In addition, the inspectors determined whether the condition and operation of the heat exchangers selected were consistent with design assumptions in heat transfer calculations, and as described in the Final Safety Analysis Report. This included determining whether the number of plugged tubes was within pre-established limits based on capacity and heat transfer assumptions. In addition, eddy current (EC) test reports and visual inspection records were reviewed for the EDG 1A1-1 HX, EDG 1A1-2 HX, CS 1A HX, CS 1B HX, and the Main Control Room A/C Chiller B-B to determine the structural integrity of the heat exchangers.

The inspectors determined whether the performance of the ultimate heat sink (UHSs) and its subcomponents such as the EDG 1A1 MOV isolation valve, traveling screens, ERCW backwash isolation valve were appropriately evaluated by tests or other equivalent methods, to ensure availability and accessibility to the in-plant cooling water systems.

The inspector performed a system walkdown of the service water (SW) intake structure to determine whether the licensee's assessment on structural integrity and component functionality was adequate, and that the licensee ensured proper functioning of traveling screens and strainers, and structural integrity of component mounts. In addition, the inspectors determined whether the SW pump bay silt accumulation was monitored, trended, and maintained at an acceptable level by the licensee, and that water level instruments were functional and routinely monitored. The inspectors also determined whether the licensee's ability to ensure functionality during adverse weather conditions was adequate.

In addition, the inspectors reviewed problem evaluation reports (PERs) and condition reports (CRs) related to the heat exchangers/coolers and heat sink performance issues to determine whether the licensee had an appropriate threshold for identifying issues, and to evaluate the effectiveness of the corrective actions. The documents that were reviewed are included in the Attachment to this report.

Documents reviewed are listed in the Attachment. These inspection activities constituted seven heat sink inspection samples as defined in inspection procedure (IP) 71111.07-05.

b. Findings

(Opened) Unresolved Item 050-390/2015002-05, Review of 50.59 Evaluation for the Emergency Diesel Generator Heat Exchanger

Introduction: The inspectors identified an unresolved item (URI) regarding the licensee's 10 CFR 50.59 evaluation for a modification to the operational configuration of the inlet motor operated valves (MOVs) for the EDG Heat Exchanger. Additional inspection would be required to determine if the licensee's 10 CFR 50.59 evaluation properly addressed whether the modification resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a structures, systems, or components (SSCs) important to safety previously evaluated in the UFSAR.

Description: Watts Bar has four EDGs that are each cooled by two heat exchangers supplied by the ERCW system. Prior to the modification, flow through the heat exchangers was continuous due to the inlet MOVs (1-FCV-067-0066-A, 2-FCV-067-0066-A, 1-FCV-067-0067-B and 22FCV-067-0067-B) being locked open. In order to ensure sufficient flow is available to components served by ERCW during dual-unit operations, the licensee modified the position of these MOVs from normally open with power removed, to normally closed with breakers closed. This resulted in the EDG heat exchangers being isolated during normal operation from the ERCW system. Flow, however, would be restored by the MOVs active function to open upon receipt of a signal from the EDG speed switch, should the EDGs startup.

The inspectors reviewed the results of the licensee's 10 CFR 50.59 evaluation related to the impact of the modification on the failure probability of the EDG. The inspectors concluded that additional information and review was necessary to determine whether the modification resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a system, structure, or component important to safety previously evaluated in the UFSAR. Particularly, the inspectors needed additional information on the specific inputs, assumptions, and evaluation methodology used to determine the increase in EDG failure probability. This issue was identified as URI 05000390/2015002-05, Review of 10 CFR 50.59 Evaluation for the EDG Heat Exchanger.

1R11 Licensed Operator Requalification and Performance (71111.11)

.1 Licensed Operator Requalification Review

a. Inspection Scope

On May 13, 2015, the inspectors observed the simulator evaluation per scenario 3-OT-SRT-INPO-CPE-4, SSPS Failure, ATWS, Small Steam Line Leak, Revision 1. The plant conditions led to a Site Area Emergency. Performance indicator credit was not taken.

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. Documents reviewed are listed in the Attachment. This activity constituted one Observation of Requalification Activity inspection sample.

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. Documents reviewed are listed in the Attachment. This activity constituted one Observation of Operator Performance inspection sample.

- 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 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. Documents reviewed are listed in the Attachment. This activity constituted one Maintenance Effectiveness inspection sample.

- Problem Evaluation Report (PER) 991105, Failure of the ERCW TCV on the A main control room chiller

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; and TI-124, Equipment to Plant Risk Matrix. Documents reviewed are listed in the Attachment. This activity constituted five Maintenance Risk Assessment inspection samples.

- Risk assessment for work week 0427 with emergent inoperability of B train control room emergency air temperature control system
- Risk assessment for work week 0504 with emergent failure of the 2B-B EDG
- Risk assessment for work week 0511 with the A ERCW header crosstied for strainer maintenance

- Risk assessment for work week 0525 with the 1-B condenser vaccuum pump
- Risk assessment for work week 0622 with the emergent inoperability of the turbine-driven auxiliary feedwater pump

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 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. Documents reviewed are listed in the Attachment. This activity constituted five Operability Evaluation inspection samples.

- Prompt determination of operability (PDO) for PER 995791, RHR FCVs are non-conforming to the ASME code
- PDO for PER 1022981, containment spray mininimum flow valve time delay less than design
- PDO for PER 1017781, 1B-B CCP surveillance run with inappropriate range on suction pressure test gauge
- PDO for PER 1022091, 2B-B EDG has a -40 VDC ground
- Operational Decision Making Instruction (ODMI) for PER 980144 1A SIP mechanical seal leakage

b. Findings

1. Scoping of Valves in In-service Testing Program

Introduction. The NRC-identified a Green NCV of 10 CFR 50.55a, “Codes and Standards,” for the licensee’s failure to meet the test requirements set forth in the American Society of Mechanical Engineers ASME Operation and Maintenance (OM) code for RHR flow control valves (FCVs). Specifically, TVA failed to scope the RHR FCVs into their IST program.

Description. RHR FCVs 1-FCV-74-16 and 1-FCV-74-28 are the A and B RHR heat exchanger outlet flow control valves, respectively. They are normally open at power and are used to throttle flow during shutdown operations to maintain adequate decay heat removal. The valves are ASME Code Class 2. They have a safety function to fail open on loss of air and they receive a safety injection signal which causes a fail open

actuation during a design bases event in order to ensure a viable emergency core cooling system (ECCS) injection flowpath.

Inspectors questioned whether the RHR FCVs should be scoped into the IST program at Watts Bar Unit 1 in late 2014. Around the same time, a vendor doing an IST review for Watts Bar Unit 2 recommended that both the Unit 1 and Unit 2 RHR FCVs be included in the IST program. However, the licensee viewed this recommendation as an enhancement and did not conclude that the valves would be required to be scoped into the IST program for Unit 1.

OM Code 2002, Subsection ISTA, “General Requirements,” Section ISTA-1100, “Scope,” states in part, “Section IST establishes the requirements for pre-service and IST and examination of certain components to assess their operational readiness in light-water reactor nuclear power plants. These requirements apply to: a) pumps and valves that are required to perform a specific function in shutting down the reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident.” 10 CFR 50.55a(f)(1), requires the establishment of OM Code IST test requirements for components which are classified ASME Code Class 1, 2 and 3. The inspectors identified that TVA had not scoped RHR FCVs in their IST program.

NUREG-1482, “Guidelines for In-service Testing at Nuclear Power Plants,” Table 2.1, “Typical Systems and Components in an In-service Testing Program for a Pressurized-Water Reactor,” includes “valves in flowpath” for residual heat removal systems. Furthermore, section 4.2.9, “Control Valves with a Safety Function,” states that control valves which are used only for system control would generally be exempt from IST. However, control valves with a safety function, e.g. fail open, must be tested in accordance with the requirements of IST to monitor the valves for degrading conditions. The inspectors determined the RHR FCVs, because they receive a safety injection signal to fail open, did have a safety function.

Inspectors noted that, since the valves had never been scoped into the IST program, required remote position indication (RPI) testing was not being performed. The licensee took corrective action to scope both RHR FCVs into the IST program and updated surveillance procedures to perform RPI testing as an IST acceptance criteria.

Analysis. The inspectors determined that the licensee’s failure to scope and meet the testing requirements of the OM Code for RHR FCVs in accordance with 10 CFR 50.55a, was a performance deficiency. The performance deficiency was more than minor because if left uncorrected, the failure to perform required IST testing could lead to a more significant safety concern in that valve degradation could go unnoticed resulting in undetected inoperability. The inspectors evaluated the significance of this finding using IMC 0609 Appendix A, dated June 19, 2012, The Significance Determination Process (SDP) for Findings at Power, Exhibit 2, Mitigating Systems Screening Questions. The inspectors 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 performance deficiency had a cross-cutting aspect of Conservative Bias in the area of Human Performance because the

licensee failed to use conservative decision making practices in their evaluation of the status of the RHR FCVs after being challenged by the NRC. (H.14)

Enforcement. 10 CFR 50.55a, "Codes and Standards," paragraph (f)(1), states, in part, that "Other pumps and valves that perform a function to shut down the reactor or maintain the reactor in a safe shutdown condition, mitigate the consequences of an accident, or provide overpressure protection for safety-related systems (in meeting the requirements of the 1986 Edition, or later, of the Boiler and Pressure Vessel or OM Code) must meet the test requirements applicable to components which are classified as ASME Code Class 2 or Class 3." Contrary to the above, since May 27, 1996, the licensee failed to meet the test requirements for Code Class 2 components because the RHR FCVs were not scoped into their IST program. Immediate corrective actions included entering this issue into their Corrective Action Program (CAP) and modifying the RHR pump testing procedures to perform the required RPI testing. Because this finding is of very low safety significance (Green) and was entered into the CAP (PER 995791), this issue is being treated as an NCV consistent with Section 2.3.2.a of the Enforcement Policy. (NCV 0500390/2015002-01: Residual Heat Removal Flow Control Valves not Scoped in In-Service Testing Program.)

2. Failure to Track Applicable TS Requirements

Introduction. The NRC identified a Green NCV of technical specification (TS) 5.7.1.1.a, Procedures, for the licensee's failure to implement OPDP-8, Operability Determinations and LCO tracking. Specifically, the licensee failed to track the applicability of action statement 'A' of TS LCO 3.5.2.A, emergency core cooling systems, during planned testing.

Description. RHR FCVs 1-FCV-74-16 and 1-FCV-74-28 are the A and B RHR heat exchanger outlet flow control valves, respectively. They are normally open at power and are used to throttle flow during shutdown operations to maintain adequate decay heat removal. The valves are ASME code class 2. They have a safety function to fail open on loss of air and they receive a safety injection signal which causes a fail open actuation during a design bases event in order to ensure a viable ECCS injection flowpath.

Inspectors identified the RHR FCVs were not scoped in the IST program and questioned the licensee on their inclusion into the program as active valves. Through discussions with the licensee the inspectors discovered a Westinghouse memo from 1992 in which TVA requested the valves be categorized as active under the IST program. Westinghouse responded that the valves could not be categorized as active for several reasons, including that certain tests necessary to demonstrate operability of the valve to open under design bases conditions were never performed. The licensee subsequently decided that the valves were not required to be scoped into the IST program at all. See NCV 0500390/2015002-01 for issues associated with the licensee's treatment of these valves under the IST program.

Inspectors identified the valves were closed at power for periods of time up to 7 hours during RHR pump performance testing and emergency core cooling system venting

surveillances inside containment. The licensee was not entering TS LCO 3.5.2.a, emergency core cooling systems for the associated RHR train during the periods of time the valve was not in its open safety position. The licensee was relying on the valves to fail open on a loss of air or a safety injection signal during testing. The residents challenged the licensee on how operability was assured during this time period in light of the Westinghouse memo which showed the valves did not meet active qualification requirements and thus cannot be relied upon to change position under accident conditions. The residents also noted the requirements of OPDP-8, "Operability Determination Process and Limiting Conditions for Operation Tracking", Rev. 18, step 3.5.1, which requires, in part, that TS LCOs for inoperable equipment be entered into the narrative logs.

The licensee took immediate corrective actions to modify procedural requirements to enter the TS LCO and log the entry in the narrative logs whenever the valves are less than full open in modes 1-3.

Analysis. The licensee's failure to track applicable technical specification action statements as required by section 3.5.1 of OPDP-8, "Operability Determination Process and Limiting Conditions for Operation Tracking" was a performance deficiency. The performance deficiency was more than minor because, if left uncorrected, it would have had the potential to lead to a more significant safety concern in that, the failure to track an applicable technical specification action statement could lead to plant operations outside of TS analyzed conditions. The inspectors evaluated the significance of this finding using IMC 0609 Appendix A, dated June 19, 2012, The Significance Determination Process (SDP) for Findings at Power, Exhibit 2, Mitigating Systems Screening Questions. The inspectors 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 and did not represent an actual loss of function of one or more non-technical specification equipment for greater than 24 hours. The performance deficiency had a cross-cutting aspect of Avoid Complacency in the area of Human Performance because licensee personnel were complacent and failed to question long held assumptions about the ability of the valves to fail to their safe position under all design basis conditions. (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 1(h), "Administrative Procedures," required procedures addressing log entries, which was partially implemented by OPDP-8, "Operability Determination Process and Limiting Conditions for Operation Tracking," Revision 18. OPDP-8, section 3.5.1, required, in part, plant log entries of entry and exit from technical specification action statements. Contrary to the above, the licensee failed to make plant log entries for the entry and exit from TS LCO 3.5.2, emergency core cooling systems condition "A" on multiple occasions prior to April 8, 2015. Immediate corrective actions included entering this issue into their CAP and modifying the RHR pump testing procedures to declare the associated RHR train inoperable when it's FCV is closed. Because this finding is of very low safety significance (Green) and was entered into the CAP (CR 1010269), this issue is being treated as an NCV consistent with Section 2.3.2.a of the Enforcement Policy.

(NCV 0500390/2015002-02: Failure to Track Applicable Technical Specification Action Statement for Residual Heat Removal System.)

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the permanent plant modification 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. Documents reviewed are listed in the Attachment. This activity constituted one Plant Modifications inspection sample.

- Design Change Notice 63886, EDG speed switch adjustment

b. Findings

No findings were identified.

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. Documents reviewed are listed in the Attachment. This activity constituted six Post-Maintenance Testing inspection samples.

- WO 16592678, cooling fan for inverter WBN-2-INV-235-001-D post-maintenance test following fan replacement
- WOs 116299911, 116811695, vital inverter 1-1 automatic transfer test following emergent circuit card replacement

- WO 116358103, MDAFW pump 1B-B performance test following maintenance outage
- WO 116714918, 2A-A EDG surveillance test following potentiometer swipe
- WO 115155935 EDG 1A-A surveillance test following 18-month maintenance outage
- WOs 116759192, 116759219, Intake pumping station train B sump pump performance testing following sump cleanout and line flushing

b. Findings

No findings were identified.

1R21 Component Design Basis Inspection (71111.21)

.1 (Closed) Unresolved Item (URI) 05000390/2015007-03, Break In Non-Seismic ERCW Discharge Piping (ML 15070A535)

a. Inspection Scope

An unresolved item (URI) was opened related to the licensee's failure to consider the effects of a break in the non-seismic portion of the Essential Raw Cooling Water (ERCW) discharge flow path to the cooling tower basin in the calculation used to determine the net positive suction head (NPSH) available to the auxiliary feedwater (AFW) pumps. The URI was to determine if the performance deficiency is More-than-Minor.

The inspectors reviewed the licensee's design and licensing basis documentation related to the ERCW system and AFW system. Additionally, the inspectors and NRC headquarters staff reviewed the licensee's prompt determination of operability. Documents reviewed are listed in the Attachment.

b. Findings

Introduction: The inspectors identified a Green non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to consider the effects of a break in the non-seismic portion of the essential raw cooling water (ERCW) discharge flow path to the cooling tower basin in the calculation used to determine the net positive suction head (NPSH) available to the auxiliary feedwater (AFW) pumps.

Description: The non-seismic normal discharge flow path of the ERCW system is to the cooling tower basin. The inspectors noted that ERCW system description document WBN-SDD-N3-67-4002, "Essential Raw Cooling Water System, System 67," Revision 0028, stated, in part, that nonsafety-related ERCW system components shall be designed such that their failures do not jeopardize safety-related components. The inspectors also noted that calculation EPMJKJ011191, "WBN AFW System – Pump Net Positive Suction Head (NPSH) Available Calculation," Revision 010, was used to determine the available NPSH for the AFW pumps.

The inspectors determined that calculation EPMJKJ011191 did not consider the effects of a break in the non-seismic portion of the discharge flow path to the cooling tower basin. A break of this type could result in a much lower backpressure on the ERCW system, and result in a reduction of available NPSH to the AFW during accident conditions. The licensee entered the issue into their CAP as PER 979323 and performed an operability determination which concluded that the AFW system would have adequate NPSH assuming a reasonable failure such as a partial break of the non-seismic piping. This determination was based in part on the similar pipe materials, installation techniques, and surrounding soil characteristics of the safety and non-safety related piping. The licensee has planned corrective actions to seismically qualify portions of the ERCW discharge path to the cooling towers, to limit the back pressure reduction if a failure occurred.

Analysis: The licensee's failure to consider the effects of a break in the non-seismic portion of the ERCW discharge flow path to the cooling tower basin in calculation EPMJKJ011101 was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, a break of the nonsafety-related ERCW discharge piping would reduce the NPSH available to the AFW pumps and adversely affected the capability of the AFW system to perform its safety function.

The inspectors used NRC Inspection Manual Chapter (IMC) 0609, Att. 4, "Initial Characterization of Findings," issued June 19, 2012, for Mitigating Systems, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component (SSC), and the SSC maintained its operability. The inspectors determined that no cross-cutting aspect was applicable because the finding was not indicative of current licensee performance.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, since March 20, 1991, the applicant failed to appropriately verify or check the adequacy of design of the ERCW system discharge piping. Specifically, the licensee failed to consider the effects of a break in the non-seismic portion of the ERCW discharge flow path to the cooling tower basin in the calculation used to determine the NPSH available to the AFW pumps. 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 PER 97923. (NCV 05000390/2015002-03, Failure To Consider The Effects Of A Break In Non-Seismic ERCW Piping)

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, American Society of Mechanical Engineers (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. Documents reviewed are listed in the Attachment. This activity constituted eight Surveillance Testing inspection samples; two in-service and six routine.

In-Service Test:

- WO 116358208, TI-50-021, Intake pumping station strainer room A sump pump performance test
- WO 116347472, 1-SI-62-901-B Centrifugal charging pump 1B-B quarterly performance test.

Routine Surveillances:

- WO116274908, 0-MI-235.002, 120 VAC vital inverter (0-II) automatic transfer test
- WO 116357732, 1-SI-72-908-A Containment spray pump 1A-A comprehensive pump test
- WO 116401639, 1-TRI-211-1, Calibration and functional tests on 6.9kV shutdown board 1A-A alternate supply breaker protective relays
- WO 116364094, 1-SI-1-108, Main steam pressure channel loop II COT
- WO 116291463, 0-SI-82-12-a, EDG 2A-A monthly start and load test
- WO 116069464, 1-SI-99-10-A, 62 day functional test of SSPS train A and reactor trip breaker A

b. Findings

Introduction: A self-revealing, Green, NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings” was identified for the licensee’s failure to follow procedure 1-SI-99-10-A, 62 Day Functional Test of SSPS Train A and Reactor Trip Breaker A, Revision 59 as amended for troubleshooting. As a result, the licensee shorted supply voltage and subsequently tripped supply breaker 1-BKR-235-3/43F which resulted in the inoperability of the 1A-A containment spray pump.

Description: On April 16, 2015, the inspectors observed the performance of 1-SI-99-10-A, 62 Day Functional Test of SSPS Train A and Reactor Trip Breaker A. The General Warning Light was lit for the SSPS train to be tested. The reason for the warning light was two blown fuses (as identified by blown fuse indicators) for the +48VDC power supply. The test was called off, and the licensee developed a troubleshooting plan. The troubleshooting plan consisted of a version of 1-SI-99-10-A which had had been revised under Procedure Control Form 070-4 to add instructions for fuse replacement under an

inserted step 18A. The residents observed that a crew member removed the fuses, and then replaced one, which blew upon installation. The troubleshooting plan step 18A did not have specific actions if the replacement fuses blew upon installation. However, a crew member and system engineer continued working and opened the rear of the affected SSPS cabinet (1-R-47). While the engineer referenced electrical drawings, the technician inserted test leads into the cabinet. One test lead was inserted into a multimeter, and the other test lead was inserted into that lead (or stacked), effectively shorting the two leads together. The technician then inserted the other ends of the leads into the cabinet thereby creating a short.

As a result of the short, the Control Room received Annunciators 19-D 120 AC VITAL PWR BD 1-III UV/CKT TRIP and 85-F RVLS SYS MALFUNCTION ALARMS. Auxiliary Unit Operators were dispatched to investigate and found 1-BKR-235-3/43 TRAIN A SSPS CAB 1-R-46 CHANNEL III INPUT in the off position. The breaker trip resulted in a loss of reactor coolant pump 3 input to the Reactor Vessel Level Indication System (RVLIS) and the licensee entered Technical Specification LCO 3.3.3 for one required channel of a Post-Accident Monitoring System (PAMS) function being inoperable. The breaker trip resulted in the inoperability of one main steam stop valve room water level channel. This channel inoperability resulted in the licensee entering TS LCO 3.3.2 for engineered safety feature actuation system (ESFAS) instrumentation. The breaker trip also resulted in entry into TS LCO 3.6.6, Condition A, one containment spray (CS) train inoperable due to a loss of flow input to the CS pump 1A-A miniflow valve.

Analysis: The failure to follow procedure 1-SI-99-10-A by attempting to take voltage measurements which were not directed by the revised procedure was a performance deficiency. The performance deficiency was more than minor because it adversely affected the equipment performance attribute of the mitigating systems cornerstone to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the failure to follow the troubleshooting procedure resulted in drawing an arc in the SSPS cabinet and tripping an upstream supply breaker which resulted in the inoperability of the 1A-A containment spray pump. The inspectors evaluated the significance of this finding using IMC 0609 Appendix A, dated June 19, 2012, The Significance Determination Process (SDP) for Findings At-Power, Exhibit 2, Mitigating Systems Screening Questions. The inspectors 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 of containment spray for greater than its Tech Spec allowed outage time. The performance deficiency had a cross-cutting aspect of Procedure Adherence in the area of Human Performance because crew members failed to follow the work instructions in the troubleshooting procedure (H8).

Enforcement: 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." Contrary to this requirement, the licensee failed to accomplish troubleshooting in accordance with procedure 1-SI-99-10-A section 6.1, step 18A, as amended, with troubleshooting instructions by procedure change request 070-4. Specifically, the licensee attempted to take voltage readings in the cabinet contrary to

the procedure which did not direct the taking of voltages if the replacement fuses blew. This resulted in a partial loss of a vital bus and affected portions of the PAMS, ESFAS, and CS system, thereby causing a loss of one train of containment spray. Because this violation was of very low safety significance (Green) and was entered into the CAP as CR 1015778, this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000390/2015002-04: Failure to Follow Procedure during SSPS Testing)

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

On April 22, 2015, the inspectors observed a licensee-evaluated emergency preparedness drill from the simulated control room 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, 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. The inspectors attended the post-drill critique to compare any inspector-observed weaknesses with those identified by the licensee in order to verify whether the licensee was properly identifying emergency preparedness (EP) related issues and entering them into the CAP, as appropriate. Documents reviewed are listed in the Attachment. This activity constituted one EP training drill inspection sample.

b. Findings

No findings were identified.

4OA1 Performance Indicator (PI) Verification (71151)

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the following Performance Indicators (PIs). The inspectors examined the licensee's PI data for the specific PIs listed below for the second quarter 2014 through first quarter of 2015. The inspectors reviewed the licensee's data and graphical representations as reported to the NRC to verify that the data was correctly reported. The inspectors validated this data against relevant licensee records (e.g., PERs, daily operator logs, plan of the day, licensee event reports [LERs], etc.), and assessed any reported problems regarding implementation of the PI program. The inspectors verified that the PI data was appropriately captured, calculated correctly, and discrepancies resolved. The inspectors used the NEI 99-02, Regulatory Assessment Performance Indicator Guideline, to ensure that industry reporting guidelines were appropriately

applied. Documents reviewed are listed in the Attachment. This activity constituted seven performance indicator inspection samples.

- Mitigating Systems Performance Indiex (MSPI) - High Pressure Injection System
- MSPI - Emergency AC Power
- MSPI - Heat Removal
- MSPI - Residual Heat Removal
- MSPI - Cooling Water MSPI
- Safety System Functional Failures
- Reactor Coolant System Leak Rate

4OA2 Problem Identification and Resolution (71152)

.1 Review of Items Entered into the CAP

a. Inspection Scope

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

b. Findings

No findings were identified

.2 Annual Sample: PER 613123 related to failure of ice condenser system glycol return header outboard containment isolation valve to remain closed during performance of local leak rate testing

a. Inspection Scope

The inspectors performed an in-depth review of the licensee's evaluation and corrective actions associated with PER 613123 related to failure of ice condenser system glycol return header outboard containment isolation valve to remain closed during performance of local leak rate testing.

On September 19, 2014, during performance of 1-SI-61-701, Containment Isolation Valve Local Leak Rate Test Ice Condenser, flow control valve 1-FCV-61-193-A, glycol return auxiliary building isolation, failed to remain closed after the handswitch was taken to the "CLOSE" position and the licensee documented this in PER 613123. Valve 1-FCV-61-193-A had a history of degraded performance with multiple WOs associated with the valve. An apparent cause evaluation (ACE) was performed for the most recent failure. Field troubleshooting activities revealed the direct cause of the valve failure to be a failure of zone switch 1-ZS-61-193A-A due to water intrusion inside its limit switch compartment. The apparent cause was determined to be a latent construction error

resulting in the associated conduit being unsealed and allowing water intrusion into the zone switch. A contributing cause was determined to be melting ice from a small section of removed insulation, exposing glycol header piping dripping into the conduit that houses the limit switch conductors. The water penetrated the conduit connections and condulet covers which yielded a pathway to inside the limit switch, causing the corrosion to the zone switch.

Immediate corrective actions were taken to replace the zone switch and install conduit sealant. Additional corrective actions were taken to reinsulate the exposed pipe and perform extent of condition walkdowns of all accessible isolation valves on or near the glycol portions of the ice condenser system susceptible to water intrusion. An additional flow control valve, 1-FCV-61-191, was identified and a WO initiated to seal the conduit and include in the next outage scheduled for fall 2015.

b. Findings and Observations

No findings were identified. The inspectors noted that the licensee took the following corrective actions under PER 613123:

- Replaced zone switch and install conduit sealant associated with 1-FCV-61-193-A
- Reinsulated exposed piping above 1-FCV-61-193-A
- Performed extent of condition (EOC) walkdowns of all accessible isolation valves on or near the glycol portions of the ice condenser system susceptible to water
- Tracked WOs generated from the EOC walkdowns to completion
- Initiated WOs for each component identified to install conduit sealant
- Reviewed outstanding insulation WOs and re-prioritize as needed
- Informed Unit 2 engineering for operating experience

The following corrective actions are still open under PER 613123:

- Initiate WO to install conduit sealant for 1-FCV-61-191, identified from EOC, to be performed at next refueling outage, fall 2015
- Track WO for installation of conduit sealant for 1-FCV-61-191 to closure

The initiating issue related to PER 613123 has been adequately resolved. However, the inspectors noted several observations related to the EOC actions related to installing conduit sealant for 1-FCV-61-191. Although the EOC was performed and the WO was issued months prior to the spring 2014 outage, it was not performed. It is unclear to the licensee why it was omitted or deleted from the outage scope. After the spring 2014 outage, the WO was then incorrectly canceled due to miscommunication between the work planners and the engineers. A subsequent WO has been reinitiated and is scheduled for the fall 2015 outage.

4OA3 Event Follow-up (71153)

(Closed) LER 05000390/2015-001, Maunal Reactor Trip Due to Rapid Loss of Main Condenser Vacuum

a. Inspection Scope

Inspectors reviewed LER 05000390/2015-001. The licensee identified that the rapid loss of condenser vacuum was caused by a failure of the C condenser expansion joint boot seal. The licensee initiated PER 991403 to enter this issue into the CAP. The root cause of seal failure was determined to be an inadequate risk assessment process for critical maintenance. This resulted in inadequate oversight of the installation vendor for the expansion joint seal. The licensee's immediate corrective actions were to replace the C expansion joint; the A and B expansion joints were also replaced as a preventative measure. The NRC inspectors reviewed processes in place at the time of the seal installation and verified there were no performance deficiencies. Inspectors also reviewed the licensee's root cause analysis and a failure analysis report prepared by a third party vendor who specializes in expansion joint boot seals. The inspectors reviewed and confirmed the specified corrective actions were appropriate and were being implemented.

b. Findings

No findings were identified.

4OA6 Meetings, including Exit

On July 30, 2015, 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.

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Arent, General Manager,WBN Site Licensing
L. Belvin, Quality Assurance Manager
M. Bottorff, Operations Superintendent
M. Casner, Director, Engineering
S. Connors, Plant Manager
T. Detchemende, Emergency Preparedness Manager
C. Dieckmann, Director, Plant Support
S. Fisher, Senior Manager, Nuclear Site Security
W. Hooks, Radiation Protection Manager
J. James, Director, Maintenance
F. Koontz, Specialist, Nuclear Engineering
D. Lee, Manager, Design Engineering
J. O'Dell, Site Licensing Supervisor
R. Proffitt, Specialist, Nuclear Construction Licensing
J. Reidy, Director, Operations
T. Sears, GL89-13 and Heat Exchanger Program Engineer
P. Stephens, Senior Manager, Chemistry
R. Stroud, Site Licensing
M. Taggart, Director, Work Management
K. Walsh, Site Vice President

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

050390/2015-002-05	URI	Review of 10 CFR 50.59 Evaluation for the Emergency Diesel Generator Heat Exchanger (1R07)

Opened and Closed

05000390/2015-002-01	NCV	Residual Heat Removal Flow Control Valves not Scoped in In-Service Testing Program. (Section 1R15.1)
05000390/2015-002-02	NCV	Failure to Track ApplicableTechnical Specification Action Statement for Residual Heat Removal System (Section 1R15.2)
05000390/2015-002-03	NCV	Failure To Consider The Effects Of A Break In Non-Seismic ERCW Piping (Section 1R21)
05000390/2015-002-04	NCV	Failure to Follow Procedure during SSPS Testing (Section 1R22)

<u>Closed</u>

05000390/2015007-03	URI	Break In Non-Seismic ERCW Piping (Section 1R21)
05000390/2015-001	LER	Manual Reactor Trip Due to Rapid Loss of Main Condenser Vaccum (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

NPG-SPP-7.1.6, On Line Work Control Power System Alerts/Offsite Power, Rev. 0004
NPG-SPP-7.1.7, Station Seasonal Readiness, Rev. 0004
TI-12.15, 161KV Offsite Power Requirements, Rev. 0027
TRO-EA-SOP-30.405, Nuclear Offsite Power Operating Requirements, Rev. 0000
TRO-EA-SOP-30.403, Nuclear Offsite Power Disqualification Notification and Call-Out Procedure, Rev. 0012
TRO-SPP-30.006, Supervisory Control Data Acquisition (SCADA) and Energy Management System (EMS) Impairment Procedure, Rev. 0008

Section 1R04: Equipment Alignment

1-SOI-62.01 Attachment 1P, CVCS Charging and Letdown Power Checklist 1-62.01-1P, Rev. 0000
1-SOI-62.01 Attachment 1V, CVCS Charging and Letdown Valve Checklist 1-62.01-1V, Rev. 0000
SR 005504
0-SOI-67.01, Essential Raw Cooling Water System, Rev. 0001, Att 4V
Drawing 1-47W845-1, Mechanical Flow Diagram – Essential Raw Cooling Water System
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LIST OF ACRONYMS

ACE	apparent cause evaluation
ADAMS	Agencywide Documents Access and Management System
AFW	Auxillary Feedwater
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CCP	centrifugal charging pump
CFR	*Code of Federal Regulations*
CS	containment spray
EBR	electric board room
EC	Eddy Cument
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOC	extent of condition
EP	Emergency Preparedness
EPIP	Emergency Plan Implimenting Procedure
ERCW	essential raw cooling water
ESFAS	engineered safety feature actuation system
FCV	Flow Control Valves
HPFP	high pressure fire pump
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IST	In-Service Testing
LCO	limiting condition for operation
LER	licensee event report
MDAFW	Motor Driven Auxillary Feedwater
MOV(s)	motor operated valves
MSPI	Mitigating Systems Performance Indiex
NCV	non-cited violation
NEI	Nuclear Energy Institute
NPG-SPP	nuclear power group standard programs and processes
NPSH	Net Position Section Head
NRC	Nuclear Regulatory Commission
ODMI	Operational Decision Making Instruction
OM	Operator and Maintenance
OOS	out of service
PAMS	Post-Accident Monitoring System
PER	problem evaluation report
PDO	prompt determination of operability
PI	performance indicator
RHR	residual heat removal
RVLIS	Reactor Vessel Level Indication System
SDP	Significance Determination Process
SSC(s)	structures, systems, or components
SSPS	Solid State Protection System
SW	Service Water

TCV	Temperature Control Valve
TS	technical specifications
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
WBN	Watts Bar Nuclear Plant
WO	work order