



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

October 21, 2015

EA-15-075

Mr. Michael D. Skaggs
Senior Vice President
WBN Operations & Construction
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

**SUBJECT: WATTS BAR NUCLEAR PLANT UNIT 2 CONSTRUCTION - NRC INTEGRATED
INSPECTION REPORT 05000391/2015608**

Dear Mr. Skaggs:

On September 30, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection of construction and testing activities at your Watts Bar Unit 2 reactor facility. The enclosed integrated inspection report documents the inspection results, which were discussed on October 15, 2015, with Ms. Cooper and other members of your staff.

This inspection examined activities conducted under your Unit 2 construction permit as they relate to safety and compliance with the Commission's rules and regulations, the conditions of your construction permit, and fulfillment of Unit 2 regulatory framework commitments. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the enclosed report documents four NRC-identified findings and one self-revealing finding which were determined to involve violations of NRC requirements. However, because the findings were all Severity Level IV violations and were entered into your corrective action program, the NRC is treating the violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the non-cited violations in the enclosed report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTENTION: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Watts Bar Unit 2 Nuclear Plant. In addition, 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 Watts Bar Unit 2 Nuclear Plant.

In accordance with 10 *Code of Federal Regulations* (CFR) 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC website at <http://www.nrc.gov/reading-rm/adams.html> (the Public Electronic Reading Room).

Should you have questions concerning this letter, please contact us.

Sincerely,

/RA/

Robert Haag, Chief
Construction Projects Branch 3
Division of Construction Projects

Docket No. 50-391
Construction Permit No: CPPR-92

Enclosure: Integrated Inspection Report 05000391/2015608
w/ Attachment

cc w/encl: (See next page)

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Letter to Michael D. Skaggs from Robert C. Haag dated October 21, 2015.

SUBJECT: WATTS BAR NUCLEAR PLANT UNIT 2 CONSTRUCTION - NRC INTEGRATED
INSPECTION REPORT 05000391/2015608

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PUBLIC

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.: 50-391

Construction Permit No.: CPPR-92

Report No.: 05000391/2015608

Applicant: Tennessee Valley Authority (TVA)

Facility: Watts Bar Nuclear Plant, Unit 2

Location: Spring City, TN 37381

Dates: August 16, 2015 – September 30, 2015

Inspectors:

- E. Patterson, Senior (Acting) Resident Inspector, Construction Projects Branch (CPB) 3, Division of Construction Projects (DCP), Region II (RII)
- J. Eargle, Resident Inspector, CPB3, DCP, RII
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Approved by:

Robert C. Haag, Chief
Construction Projects Branch 3
Division of Construction Projects

SUMMARY

Watts Bar Nuclear Plant, Unit 2

This integrated inspection included aspects of engineering and construction activities performed by Tennessee Valley Authority (TVA) associated with the Watts Bar Nuclear (WBN) Plant Unit 2 construction project. This report covered a seven-week period of inspections in the areas of quality assurance (QA), identification and resolution of construction problems, engineering and construction activities, preoperational testing, and follow-up of other activities. The inspection program for Unit 2 construction activities is described in Nuclear Regulatory Commission (NRC) Inspection Manual Chapter (IMC) 2517, "Watts Bar Unit 2 Construction Inspection Program." Information regarding the WBN Unit 2 Construction Project and NRC inspections can be found at <http://www.nrc.gov/info-finder/reactor/wb/watts-bar.html>.

Inspection Results

- The NRC identified Severity Level a (SL) IV non-cited violation (NCV) of 10 *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion XII, "Control of Measuring and Test Equipment," for the failure to establish measures to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality were properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits. Specifically, the applicant failed to perform calibrations on a specified interval for digital stopwatches that were used for activities affecting quality. The inspectors determined that the applicant's failure to perform calibrations at a specified interval was a performance deficiency. This performance deficiency was considered to be more than minor because the finding represents an inadequate process, procedure, or quality oversight function, that if left uncorrected, could adversely affect the quality of the fabrication, construction, testing, analysis, or records of a safety-related SSC, in accordance with Appendix C, Minor Violations and Findings, of IMC 2517, "Watts Bar Unit 2 Construction Inspection Program," dated June 6, 2013. This issue was entered into the corrective action program as condition report (CR) 1083002. The inspectors reviewed this finding against cross-cutting area components as described in IMC 0310, "Components Within the Cross-Cutting Areas," and determined that no cross-cutting aspect applied. (Section P.1.1)
- The NRC identified a SL IV NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The inspectors identified two examples where the applicant failed to prescribe documented instructions that were appropriate to the circumstance for testing to demonstrate operability of safety-related structures, systems, and components (SSCs). Specifically, performance of a surveillance test instruction resulted in unacceptable preconditioning of the containment air return (CAR) backdraft damper prior to recording the force required to open the backdraft damper. Additionally, surveillance test instructions resulted in unacceptable preconditioning of the hydrogen igniter circuits prior to recording the required current or temperature output. The inspectors determined that the applicant's failure to establish adequate surveillance test instructions, that caused unacceptable preconditioning of SSCs, was a performance deficiency. The inspectors determined that the performance deficiency was more than minor because it represented an inadequate process or procedure that if left uncorrected, could adversely affect testing of safety-related SSCs, in accordance with Appendix C, Minor Violations and Findings, of IMC 2517, "Watts Bar Unit 2 Construction Inspection Program," dated June 6, 2013. In one example, this condition could give a false "as-left" torque value of the CAR backdraft damper, as required by the TS

surveillance acceptance criteria. For the hydrogen igniter circuits, this condition could give a false “as-found” amperage value in the circuits and could affect the measured temperature of the hydrogen igniter elements, as required by the TS surveillance acceptance criteria. Performing the surveillances as written could mask the conditions used to determine equipment operability. These issues were entered into the corrective action program as CR 1083315 and CR 1088279. The inspectors reviewed this finding against cross-cutting area components as described in IMC 0310, “Components Within the Cross-Cutting Areas,” and determined that no cross-cutting aspect applied. (Section P.1.2)

- A self-revealing, SL IV, NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified when certain loads failed to properly strip from their electrical bus during Loss of Offsite Power Testing. Upon review it was discovered that maintenance had been performed on part of the circuitry and that the applicant failed to provide clear directions for work scope or specify a rigorous post modification test to ensure that all affected equipment was returned to an operable status. The applicant’s failure to include adequate post modification testing to verify circuit integrity in the approved work instructions was a performance deficiency. The inspectors determined that the performance deficiency was more than minor because it represented an inadequate process or procedure that if left uncorrected, could adversely affect testing of safety-related SSCs, in accordance with Appendix C, Minor Violations and Findings, of IMC 2517, “Watts Bar Unit 2 Construction Inspection Program,” dated June 6, 2013. This issue was entered into the applicant’s corrective action program as CR 1081117. The inspectors reviewed this finding against cross-cutting area components as described in IMC 0310, “Components Within the Cross-Cutting Areas,” and determined that no cross-cutting aspect applied. (Section P.1.11)
- The NRC identified a SL IV NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the applicant’s failure to provide a design drawing that assured that main steam line radiation monitor 2-RE-90-421 would be capable of performing its intended design basis function. The inspectors determined that the applicant’s failure to provide a design drawing that assured that main steam line radiation monitor 2-RE-90-421 would be capable of performing its intended design basis function was a performance deficiency. The inspectors determined the performance deficiency was more than minor in accordance with Appendix C, Minor Violations and Findings, of IMC 2517, because it represented an improper work practice that included failures to establish or implement adequate procedure functions. If left uncorrected, the installed configuration could adversely impact the function of components required for post-accident monitoring of effluent release paths. This issue was entered into the corrective action program as CR 1078327. The finding has a cross-cutting aspect in the Documentation component of the Human Performance area, as defined in IMC 0310, because design drawings were not accurate [H.7]. (Section OA.1.4)
- The NRC identified a SL IV NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the applicant’s failure to install main steam line radiation monitors 2-RE-90-422 and 2-RE-90-423 according to their design drawing. The inspectors determined that the applicant’s failure to perform installations in accordance with design drawings in order to ensure that main steam line radiation monitors 2-RE-90-422 and 2-RE-90-423 would be capable of performing their intended design basis function was a performance deficiency. The inspectors determined the performance deficiency was more than minor in accordance with Appendix C, Minor

Violations and Findings, of IMC 2517, because it represented an improper work practice that included failures to establish or implement adequate procedure functions. If left uncorrected, the installed configuration could adversely impact the function of components required for post-accident monitoring of effluent release paths. This issue was entered into the corrective action program as CR 1078327. The inspectors reviewed this finding against cross-cutting area components as described in IMC 0310, "Components Within the Cross-Cutting Areas," and determined that no cross-cutting aspect applied. (Section OA.1.4)

- The inspectors concluded that issues pertaining to several open items, including seven inspection procedures (IPs), three temporary instructions (TI), two construction deficiency reports (CDRs), two supplemental safety evaluation report (SSER) Appendix HH items, one unresolved item (URI), two generic letter (GL) items, three bulletins (BL), two Three Mile Island (TMI) action items, and one enforcement action (EA) have been appropriately addressed for WBN Unit 2.
- Other areas inspected were adequate with no findings identified. These areas included QA; operations; mechanical components; preoperational testing activities; and various NRC inspection procedures.

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REPORT DETAILS

Summary of Plant Status

During the inspection period covered by this report, Tennessee Valley Authority (TVA) performed construction completion and preoperational testing activities on safety-related systems and continued engineering design activities of the Watts Bar Nuclear (WBN) Plant, Unit 2 (U2).

I. QUALITY ASSURANCE PROGRAM

Q.1.1 Identification and Resolution of Construction Problems (Inspection Procedure 35007)

a. Inspection Scope

The inspectors continued to review condition reports (CRs), as part of the applicant's corrective action program, to verify that issues being identified under the corrective action program were being properly identified, addressed, and resolved by the applicant.

The inspectors reviewed corrective actions for CR 1078083, "Hardware nonconformance for pipe support 03B2AFWR159." The support exceeded the required strut angle of 1.5 degrees as shown on the design drawings. The inspectors reviewed the engineering evaluation and interviewed two design engineers to verify that the corrective actions were complete and the engineering evaluation was adequate and demonstrated that the support would meet its safety function. The inspectors also reviewed corrective actions for CR 912264, material traceability for threaded rod. The inspectors reviewed the material traceability of the threaded rod and visually inspected in the field that the replacement rod was not damaged.

The inspectors also reviewed and followed up on the corrective actions of several CRs discussed throughout various sections of this report.

b. Observations and Findings

No findings were identified.

c. Conclusions

The issues identified in the CRs reviewed were adequately identified, addressed, and resolved.

Q.1.2 Safety Conscious Work Environment (Inspection Procedure 35007 and Temporary Instruction 2512/015)

a. Inspection Scope

The inspectors met with TVA employee concerns program coordinators to discuss a sample of employee concerns. The inspectors reviewed existing program requirements and recent concerns identified by the applicant's and contractor's employee concerns programs.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors did not identify any issues or concerns regarding the ability of the applicant to provide a safety-conscience work environment.

II. **MANAGEMENT OVERSIGHT AND CONTROLS**

C.1 **Construction Activities**

C.1.1 **Unit 1 and Unit 2 Construction Activity Interface Controls**

a. Inspection Scope

The inspectors independently assessed applicant controls, associated with Unit 2 construction work activities, to prevent adverse impact on Unit 1 operational safety. The inspectors attended routine Unit 1/Unit 2 interface meetings to assess the exchange and sharing of information between the two site organizations. Periodic construction and planning meetings were observed, at least once per week, to assess the adequacy of the applicant's efforts to identify those construction activities that could potentially impact the operating unit. This included the review of select work activities, which the applicant had screened as not affecting Unit 1, to verify the adequacy of that screening effort. Additionally, the inspectors independently assessed select construction activities to verify that potential impacts on the operating unit had been identified and adequately characterized with appropriate management strategies planned for implementation. Furthermore, the inspectors performed independent walkdowns of select construction work locations to verify that controls to protect the operating unit provided an adequate level of protection and had been properly implemented.

Specific work activity observed included work associated with:

- 2-PTI-67-04, ERCW Three Pump Test, Revision (Rev.) 0001;
- T-Mod WBN-0-2015-030-001, ABSCE Boundary Extension to Include the Unit 2 Reactor Building;
- Work order (WO) 116923934, Collect CCS Train B Flow Data with Both CCS Pumps;
- 2-PTI-262-02, Unit 2 Integrated Safeguards Test – Train 2B, Rev. 0001; and
- 2-PTI-262-01, Unit 2 Integrated Safeguards Test – Train 2A, Rev. 0001.

Specific work activities that the applicant had screened out as not affecting Unit 1 included, but were not limited to, work activities as noted in this inspection report.

b. Observations and Findings

No findings were identified.

c. Conclusions

Overall, management oversight and controls were in place for observed construction activities that could potentially impact the operating unit.

C.1.2 (Closed) Reactor Vessel and Internals Work Observation (Inspection Procedure 50053)

a. Inspection Scope

Background: As described in inspection manual chapter (IMC) 2517, TVA addressed WBN Unit 1 construction quality issues as part of the implementation of its Nuclear Performance Plan (NPP). The results of the NRC inspection program were published in NUREG-1528, "Reconstitution of the IMC 2512 Construction Inspection Program for Watts Bar Unit 1." In 1985, construction on Watts Bar Unit 1 and Watts Bar Unit 2 was stopped due to the identification of multiple construction quality assurance (QA) issues. TVA completed Unit 1 in 1995 but had conducted very little Unit 2-specific work since 1985. In 2007, TVA decided to finish the Unit 2 plant. As part of confirming that all issues and inspection requirements will be completed for Unit 2, a review of all NRC inspection reports was initiated to determine the status of the required inspection procedures (IPs), contained in NRC IMC 2512, in effect at the time construction was stopped. This effort was called the reconstitution process. The NRC used the results of the reconstitution process to identify areas which require additional inspections. Integrated Inspection Report (IIR) 05000391/2009602 (Agencywide Documents Access and Management System (ADAMS) Accession Number (No.) ML091210420), Attachment 2, documented the reconstitution results for IP 50053. As stated in the IIR, the requirements of IP 50053 had been previously met. However, since long dormancy and newly established storage/protection conditions of the reactor pressure vessel (RPV) and internals would require additional inspections, the IIR stated that Sections 02.03b, Installed Vessel Protection, and 02.03d, Installed Vessel Internals, would continue to be performed.

Inspection Activities: The purpose of IP 50053 is to determine by direct observation and independent evaluation of work performance, work in progress, and completed work, whether activities relative to the reactor vessels and internals are being accomplished in accordance with NRC requirements and safety analysis report (SAR) commitments.

The reconstitution effort, documented in IIR 05000391/2009602, Attachment 2, concluded all requirements for this IP had been previously met. At the discretion of regional management, additional inspections were conducted for Sections 02.03.b and 02.03.d of this IP. The inspections required to be conducted, as well as the additional inspections performed during the construction phase of Watts Bar Unit 2, are documented in the following inspection reports:

- 05000391/2009602 Section C.1.2 (ADAMS Accession No. ML091210420)
- 05000391/2009603 Section C.1.2 (ADAMS Accession No. ML092120326)

- 05000391/2009604 Section C.1.3 (ADAMS Accession No. ML093030479)
- 05000391/2009605 Section C.1.1 (ADAMS Accession No. ML100290703)
- 05000391/2010602 Section C.1.2 (ADAMS Accession No. ML101230144)
- 05000391/2010603 Section C.1.7 (ADAMS Accession No. ML102170465)
- 05000391/2010604 Section C.1.9 (ADAMS Accession No. ML103060240)
- 05000391/2010605 Section C.1.10 (ADAMS Accession No. ML110410680)
- 05000391/2011605 Section C.1.2 (ADAMS Accession No. ML112201418)
- 05000391/2011607 Section C.1.2 (ADAMS Accession No. ML112730197)
- 05000391/2011608 Section C.1.2 (ADAMS Accession No. ML11311A082)
- 05000391/2011610 Section C.1.5 (ADAMS Accession No. ML12034A202)
- 05000391/2012604 Section C.1.2 (ADAMS Accession No. ML12167A212)
- 05000391/2012608 Section C.1.6 (ADAMS Accession No. ML12319A368)
- 05000391/2012609 Section C.1.5 (ADAMS Accession No. ML12356A073)
- 05000391/2012610 Section C.1.6 (ADAMS Accession No. ML13035A201)
- 05000391/2013605 Section C.1.3 (ADAMS Accession No. ML13220A640)
- 05000391/2013608 Section C.1.4 (ADAMS Accession No. ML13316A776)
- 05000391/2013610 Section C.1.5 (ADAMS Accession No. ML14049A158)
- 05000391/2013612 Section C.1.5 (ADAMS Accession No. ML13088A066)
- 05000391/2014603 Section C.1.6 (ADAMS Accession No. ML14129A381)
- 05000391/2014604 Section C.1.4 (ADAMS Accession No. ML14177A214)
- 05000391/2014607 Section C.1.4 (ADAMS Accession No. ML14274A076)
- 05000391/2014614 Section C.1.4 (ADAMS Accession No. ML14363A315)
- 05000391/2015603 Section C.1.2 (ADAMS Accession No. ML15124A921)
- 05000391/2015604 Section C.1.2 (ADAMS Accession No. ML15181A446)

Section 02.01.a of this IP required that the inspectors, as it relates to the protection of the stored vessel, (1) ascertain whether the reactor vessel is being stored in accordance with approved procedures, (2) observe seals and devices to identify vessel internal atmospheric conditions and verify maintenance of requirements, (3) observe condition of protective coating and/or protective covering and verify conformance with requirements, and (4) observe support structures and determine whether protection from entry of dirt, water, flooding (height of structure), and strength of support (shifting or collapse of structure) is adequate and consistent with storage specifications. The requirements of this section were previously met and are documented in the IIR 05000391/2009602, Attachment 2. No additional inspections beyond that IIR were performed under this inspection attribute.

Section 02.01.b of this IP required that the inspectors, as it relates to the installation techniques of the RPV and the internals, (1) ascertain whether lifting and handling activities are consistent with established specifications and procedures, including special requirements and precautions, (2) ascertain whether lifting equipment is as specified, and the required lift tests and/or nondestructive examination (NDE) of equipment have been satisfactorily accomplished, and (3) observe whether placement is being, or has been, accomplished as specified in the areas of lifting and accessory equipment, route used, and rate of vessel travel during placement. Additionally the IP required that the inspector observe the installed reactor vessel and, based on a review of the installation

specifications, drawings, and work procedures, determine whether work procedures have been or are being followed, installation is in accordance with specifications and procedures, and quality control (QC) inspection procedures are being followed. The requirements of this section were previously met and are documented in IIR 05000391/2009602, Attachment 2. No additional inspections beyond that IIR were performed under this inspection attribute.

Section 02.01.c of this IP required that the inspectors, as it relates to the protection of the installed vessel, (1) ascertain whether procedures for protection of the installed reactor vessel were followed, (2) ascertain whether protective devices are installed around the top of the open vessel to prevent entry of foreign objects and debris, (3) observe whether vessel side openings are blanked off to prevent entry of foreign objects and nozzle ends are protected from construction damage, (4) observe whether work platforms and scaffolding inside vessel are nonflammable or treated to prevent spread of accidental fires, and (5) determine whether access control activities meet requirements. The requirements of this section were previously met and are documented in the IIR 05000391/2009602, Attachment 2. Additional inspections beyond that IIR were performed under this inspection attribute and are documented in the following inspection reports:

- 05000391/2009602 Section C.1.2 (ADAMS Accession No. ML091210420)
- 05000391/2009603 Section C.1.2 (ADAMS Accession No. ML092120326)
- 05000391/2009604 Section C.1.3 (ADAMS Accession No. ML093030479)
- 05000391/2009605 Section C.1.1 (ADAMS Accession No. ML100290703)
- 05000391/2010602 Section C.1.2 (ADAMS Accession No. ML101230144)
- 05000391/2010603 Section C.1.7 (ADAMS Accession No. ML102170465)
- 05000391/2010604 Section C.1.9 (ADAMS Accession No. ML103060240)
- 05000391/2010605 Section C.1.10 (ADAMS Accession No. ML110410680)
- 05000391/2011605 Section C.1.2 (ADAMS Accession No. ML112201418)
- 05000391/2011607 Section C.1.2 (ADAMS Accession No. ML112730197)
- 05000391/2011608 Section C.1.2 (ADAMS Accession No. ML11311A082)
- 05000391/2012604 Section C.1.2 (ADAMS Accession No. ML12167A212)
- 05000391/2013612 Section C.1.5 (ADAMS Accession No. ML13088A066)
- 05000391/2013605 Section C.1.3 (ADAMS Accession No. ML13220A640)
- 05000391/2013608 Section C.1.4 (ADAMS Accession No. ML13316A776)
- 05000391/2013610 Section C.1.5 (ADAMS Accession No. ML14049A158)
- 05000391/2014603 Section C.1.6 (ADAMS Accession No. ML14129A381)
- 05000391/2014607 Section C.1.4 (ADAMS Accession No. ML14274A076)
- 05000391/2012610 Section C.1.6 (ADAMS Accession No. ML13035A201)

Section 02.02.a of this IP required inspectors to observe the storage, protection/protective coverings of the vessel internals, and the storage of supports for the stored vessel internals, and ascertain whether requirements of applicable specifications, work, and inspection procedures are being met. The requirements of this section were previously met and are documented in the IIR 05000391/2009602, Attachment 2. Additional inspections beyond that IIR were performed under this inspection attribute and are documented in the following inspection reports:

- 05000391/2009604 Section C.1.3 (ADAMS Accession No. ML093030479)
- 05000391/2009605 Section C.1.1 (ADAMS Accession No. ML100290703)
- 05000391/2010602 Section C.1.2 (ADAMS Accession No. ML101230144)
- 05000391/2010603 Section C.1.7 (ADAMS Accession No. ML102170465)
- 05000391/2010604 Section C.1.9 (ADAMS Accession No. ML103060240)
- 05000391/2010605 Section C.1.10 (ADAMS Accession No. ML110410680)
- 05000391/2012604 Section C.1.2 (ADAMS Accession No. ML12167A212)
- 05000391/2012608 Section C.1.6 (ADAMS Accession No. ML12319A368)
- 05000391/2013612 Section C.1.5 (ADAMS Accession No. ML13088A066)
- 05000391/2013605 Section C.1.3 (ADAMS Accession No. ML13220A640)
- 05000391/2013608 Section C.1.4 (ADAMS Accession No. ML13316A776)
- 05000391/2013610 Section C.1.5 (ADAMS Accession No. ML14049A158)
- 05000391/2014604 Section C.1.4 (ADAMS Accession No. ML14177A214)
- 05000391/2012610 Section C.1.6 (ADAMS Accession No. ML13035A201)

Section 02.02.b of this IP required inspectors to observe and/or verify whether the (1) lifting and handling of vessel internals are consistent with established requirements and precautions, (2) lifting equipment is as specified and required testing has been completed prior to lifting, (3) placement has been accomplished in conformance with requirements, and (4) the installation has been accomplished in conformance with requirements, including special requirements and precautions to ascertain whether work procedures have been followed, installation is in accordance with specifications, and QC inspection activities are in accordance with established procedures. The requirements of this section were previously met and are documented in the IIR 05000391/2009602, Attachment 2. Additional inspections beyond that IIR were performed under this inspection attribute and are documented in the following inspection reports:

- 05000391/2014604 Section C.1.4 (ADAMS Accession No. ML14177A214)
- 05000391/2014607 Section C.1.4 (ADAMS Accession No. ML14274A076)
- 05000391/2012609 Section C.1.5 (ADAMS Accession No. ML12356A073)

Section 02.03.a of this IP required inspectors to (1) determine whether the reactor vessel is being stored in accordance with approved procedures, (2) observe seals and devices to identify vessel internal atmospheric conditions and verify maintenance or requirements, (3) observe condition of protective coating and/or protective covering and verify conformance with requirements, and (4) observe support structures and determine whether protection from entry of dirt, water, flooding (height of structure) and strength of support (shifting or collapse of structure) is adequate and consistent with storage specifications. The requirements of this section were previously met and are documented in the IIR 05000391/2009602, Attachment 2. No additional inspections beyond that IIR were performed under this inspection attribute.

Section 02.03.b of this IP required the inspectors to (1) verify that procedures for protection of the installed reactor vessel are adequate and are being followed, (2) verify that protective devices are installed around the top of open vessel to prevent entry of foreign objects and debris, (3) observe whether vessel side openings are blanked off to prevent entry of foreign objects and nozzle ends are protected from construction

damage, (4) observe whether work platforms and scaffolding inside vessel are nonflammable or treated to prevent spread of accidental fires, and (5) determine whether access control activities meet requirements. The requirements of this section were previously met and are documented in the IIR 05000391/2009602, Attachment 2. However, the IIR stated that the long dormancy and newly established storage/protection conditions of the RPV and internals would require additional inspections. As a result, this inspection attribute was required to continue and the additional inspections beyond that IIR are documented in the following inspection reports:

- 05000391/2009602 Section C.1.2 (ADAMS Accession No. ML091210420)
- 05000391/2009603 Section C.1.2 (ADAMS Accession No. ML092120326)
- 05000391/2009604 Section C.1.3 (ADAMS Accession No. ML093030479)
- 05000391/2009605 Section C.1.1 (ADAMS Accession No. ML100290703)
- 05000391/2010602 Section C.1.2 (ADAMS Accession No. ML101230144)
- 05000391/2010603 Section C.1.7 (ADAMS Accession No. ML102170465)
- 05000391/2010604 Section C.1.9 (ADAMS Accession No. ML103060240)
- 05000391/2010605 Section C.1.10 (ADAMS Accession No. ML110410680)
- 05000391/2011605 Section C.1.2 (ADAMS Accession No. ML112201418)
- 05000391/2011607 Section C.1.2 (ADAMS Accession No. ML112730197)
- 05000391/2011608 Section C.1.2 (ADAMS Accession No. ML11311A082)
- 05000391/2011610 Section C.1.5 (ADAMS Accession No. ML12034A202)
- 05000391/2012604 Section C.1.2 (ADAMS Accession No. ML12167A212)
- 05000391/2012608 Section C.1.6 (ADAMS Accession No. ML12319A368)
- 05000391/2013612 Section C.1.5 (ADAMS Accession No. ML13088A066)
- 05000391/2013605 Section C.1.3 (ADAMS Accession No. ML13220A640)
- 05000391/2013608 Section C.1.4 (ADAMS Accession No. ML13316A776)
- 05000391/2013610 Section C.1.5 (ADAMS Accession No. ML14049A158)
- 05000391/2014603 Section C.1.6 (ADAMS Accession No. ML14129A381)
- 05000391/2014604 Section C.1.4 (ADAMS Accession No. ML14177A214)
- 05000391/2014607 Section C.1.4 (ADAMS Accession No. ML14274A076)
- 05000391/2012610 Section C.1.6 (ADAMS Accession No. ML13035A201)

Section 02.03.c of this IP required inspectors to determine whether (1) internals storage requirements are being adhered to, (2) protection/protective coverings are as specified, (3) storage supports are adequate, and (4) inspection procedures are being followed. The requirements of this section were previously met and are documented in the IIR 05000391/2009602, Attachment 2. Additional inspections beyond that IIR were performed under this inspection attribute and are documented in the following inspection reports:

- 05000391/2009603 Section C.1.2 (ADAMS Accession No. ML092120326)
- 05000391/2009604 Section C.1.3 (ADAMS Accession No. ML093030479)
- 05000391/2009605 Section C.1.1 (ADAMS Accession No. ML100290703)
- 05000391/2010602 Section C.1.2 (ADAMS Accession No. ML101230144)
- 05000391/2010603 Section C.1.7 (ADAMS Accession No. ML102170465)

- 05000391/2010604 Section C.1.9 (ADAMS Accession No. ML103060240)
- 05000391/2010605 Section C.1.10 (ADAMS Accession No. ML110410680)
- 05000391/2012604 Section C.1.2 (ADAMS Accession No. ML12167A212)
- 05000391/2012608 Section C.1.6 (ADAMS Accession No. ML12319A368)
- 05000391/2013605 Section C.1.3 (ADAMS Accession No. ML13220A640)
- 05000391/2013608 Section C.1.4 (ADAMS Accession No. ML13316A776)
- 05000391/2013610 Section C.1.5 (ADAMS Accession No. ML14049A158)
- 05000391/2014604 Section C.1.4 (ADAMS Accession No. ML14177A214)
- 05000391/2014607 Section C.1.4 (ADAMS Accession No. ML14274A076)
- 05000391/2012610 Section C.1.6 (ADAMS Accession No. ML13035A201)

Section 02.03.d of this IP required inspectors, upon the phases of construction completion, to (1) determine whether procedures for protection of the installed vessel internals are being followed, (2) observe whether protective devices are installed around top of open vessel to prevent entry of foreign objects and debris into the vessel internals, (3) observe whether vessel side openings are blanked off to prevent entry of foreign objects, (4) determine whether access control is provided to assure only authorized entry of tools, equipment, and personnel, and (5) determine whether cleanliness requirements are being met. The requirements of this section were previously met and are documented in the IIR 05000391/2009602, Attachment 2. However, the IIR stated that the long dormancy and newly established storage/protection conditions of the RPV and internals would require additional inspections. As a result, this inspection attribute was required to continue and the additional inspections beyond that IIR are documented in the following inspection reports:

- 05000391/2009602 Section C.1.2 (ADAMS Accession No. ML091210420)
- 05000391/2009603 Section C.1.2 (ADAMS Accession No. ML092120326)
- 05000391/2011605 Section C.1.2 (ADAMS Accession No. ML112201418)
- 05000391/2011607 Section C.1.2 (ADAMS Accession No. ML112730197)
- 05000391/2013612 Section C.1.5 (ADAMS Accession No. ML13088A066)
- 05000391/2014607 Section C.1.4 (ADAMS Accession No. ML14274A076)
- 05000391/2014614 Section C.1.4 (ADAMS Accession No. ML14363A315)

During this period, the inspectors conducted inspections of the RPV and reactor vessel internals storage, preservation, housekeeping, and protection activities to determine whether requirements, work procedures, and QC inspection procedures were being met. These activities are controlled by procedure 25402-000-GPP-0000-N2102, "Housekeeping," Rev. 9. The RPV was inspected to verify that the RPV was protected from foreign objects, the RPV was protected from construction damage, and that cleanliness requirements were being met. The core barrel and internals were inspected to verify storage locations were adequate and controls were in place to protect from construction damage.

In addition, the inspectors observed lifting activities of the reactor vessel lower internals, (i.e., the core barrel). The lifting and handling activities were observed to verify the activity was completed in accordance with approved work procedures WO 116225719, "Disassemble/Reassemble the U2 Reactor Pressure Vessel IAW 2-MI-68.001," and WO 117001713, "Remove and Install Lower Internals per 2-MI-058.003." The inspectors

also reviewed a sample of installation records to verify the lower internals were installed, positioned, and leveled in accordance with procedure 2-MI-68.003, "Removal and Replacement of the Unit 2 Reactor Vessel Lower Internals," Rev. 0. After the lower internals were installed in the final location, the inspectors inspected the lower internals to verify the procedures to protect the internals were being followed. The inspectors observed the protection and cleanliness activities for the installed lower internals to verify preservation, housekeeping, and protection activities were met in accordance with the work procedures, and QC inspection procedures.

The following samples were inspected:

- IP 50053 Section 02.01.c - one sample
- IP 50053 Section 02.02.a - one sample
- IP 50053 Section 02.02.b - one sample
- IP 50053 Section 02.03.b - one sample
- IP 50053 Section 02.03.c – one sample
- IP 50053 Section 02.03.d – one sample

b. Observations and Findings

No findings were identified.

c. Conclusions

The RPV and internals, Installation and Storage inspection procedure 50053, previously met all the requirements as outlined in IIR 05000391/2009602, Attachment 2. The long dormancy and newly established storage/protection conditions of the RPV and internals required additional inspections under this procedure. The inspectors observed the final placement of the RPV and lower internals and verified all procedure and devices required for the protection of these structures have been implemented. As a result, IP 50053 is considered closed; however, additional inspections may be performed at the NRC's discretion, to include when the upper internals are installed to support fuel loading activities.

C.1.3 (Closed) Reactor Vessel and Internals Record Review (Inspection Procedure 50055)

a. Inspection Scope

Background: As discussed in Section C.1.2, the reconstitution process was documented in IIR 05000391/2009602, Attachment 2, and included the results for IP 50055. As stated in the IIR the requirements of IP 50055 had been previously met. However, since long dormancy and newly established storage/protection conditions of the RPV and internals would require additional inspections. The IIR stated that Sections 02.02.b, Storage and Installation Inspections, except for the vessel installation; and 02.02.c, Nonconformance/Deviation Records, if available, will continue to be performed.

Inspection Activities: The purpose of this IP is to, by a sample review of reactor vessel and internals records, determine whether (1) the applicant system for preparing, reviewing, and maintaining records is functioning properly, (2) the selected records

reflect work accomplishment consistent with NRC requirements and SAR commitments, and (3) the records indicate any potentially generic problems, management control inadequacies, or other weaknesses that could have safety significance.

The reconstitution effort, documented in IIR 05000391/2009602, Attachment 2, concluded all requirements for this IP had been previously met. The inspection reports documenting these samples are listed within that IIR. Despite the inspection requirements having been previously met, the long dormancy and newly established storage/protection conditions of the RPV and internals required additional inspections. These additional inspections are covered under Sections 02.02.b, Storage and Installation Inspections, except for the vessel installation; and 02.02.c, Nonconformance/Deviation Records, if available. Additionally, at the discretion of regional management, additional inspections were conducted in inspection areas covered by this IP. The inspections required to be conducted per the IIR, as well as the additional inspections performed during the construction phase of Watts Bar Unit 2, are documented in the following inspection reports:

- 05000391/2009602 Section C.1.2 (ADAMS Accession No. ML091210420)
- 05000391/2009605 Section C.1.1 (ADAMS Accession No. ML100290703)
- 05000391/2012604 Section C.1.8 (ADAMS Accession No. ML12167A212)
- 05000391/2013605 Section C.1.3 (ADAMS Accession No. ML13220A640).

Section 02.01 of this IP required inspectors to review applicant/contractor requirements covering the span of records for reactor vessel and internals installation. In addition, the inspector was required to determine who prepares each record, and who is required to review the records for accuracy and for assuring that the recorded information meets requirements. The requirements of this section were previously met and are documented in the IIR 05000391/2009602, Attachment 2. No additional inspections beyond that IIR were performed under this inspection attribute.

Section 02.02.a of this IP required the inspector to review a sample of records and determine whether the applicant/contractor system for documenting safety-related work in the area of receipt inspection and material certification is functioning properly. The inspector is required to review records to confirm that the required material characteristics performance tests, non-destructive tests, environmental qualification tests, and other specification requirements were met and to ascertain that the original records or certification system met requirements of applicable criteria. The requirements of this section were previously met and are documented in IIR 05000391/2009602, Attachment 2. No additional inspections beyond that IIR were performed under this inspection attribute.

Section 02.02.b of this IP required the inspector to review a sample of records and determine whether the applicant/contractor system for documenting safety-related work in the area of storage and installation inspections is functioning properly. The inspector was required to verify that records confirm stored equipment inspections were made at the required frequency, verify that records confirm specified components were installed as specified, verify that records confirm the required inspections were performed, and verify that records confirm required protection was provided after installation. The requirements of this section were previously met and are documented in IIR 05000391/2009602, Attachment 2. However, the IIR stated that the long dormancy and

newly established storage/protection conditions of the RPV and internals would require additional inspections. As a result, this inspection attribute was required to continue and the additional inspections beyond that IIR are documented in the following inspection reports:

- 05000391/2009602 Section C.1.2 (ADAMS Accession No. ML091210420)
- 05000391/2009605 Section C.1.1 (ADAMS Accession No. ML100290703)
- 05000391/2012604 Section C.1.8 (ADAMS Accession No. ML12167A212)

Section 02.02.c of this IP required the inspector to review a sample of records and determine whether the applicant/contractor system for documenting safety-related work in the area of nonconformance/deviation records is functioning properly. The inspector was required to verify (1) records include current status of these items; (2) records are legible, complete, reviewed by QC personnel, and readily retrievable; and (3) nonconformance reports include status of corrective action or resolution. The requirements of this section were previously met and are documented in IIR 05000391/2009602, Attachment 2. However, the IIR stated that the long dormancy and newly established storage/protection conditions of the RPV and internals would require additional inspections. As a result, this inspection attribute was required to continue and the additional inspections beyond that IIR are documented in the following inspection reports:

- 05000391/2009602 Section C.1.2 (ADAMS Accession No. ML091210420)
- 05000391/2012604 Section C.1.8 (ADAMS Accession No. ML12167A212)
- 05000391/2013605 Section C.1.3 (ADAMS Accession No. ML13220A640)

Section 02.02.d of this IP required the inspector to review a sample of records and determine whether the applicant/contractor system for documenting safety-related work in the area of qualification records of craft, QA and QC inspection personnel is functioning properly. The inspector is required to verify records are complete and current and verify that records establish that personnel are adequately qualified for their assigned duties and responsibilities. The requirements of this section were previously met and are documented in IIR 05000391/2009602, Attachment 2. Additional inspections beyond that IIR were performed under this inspection attribute and are documented in the following inspection report:

- 05000391/2012604 Section C.1.8 (ADAMS Accession No. ML12167A212)

Section 02.02.e of this IP required the inspector to review a sample of records and determine whether the applicant/contractor system for documenting safety-related work in the area of QA Audits is functioning properly. The inspector is required to verify records establish that the required audits were performed and to verify deficiencies identified during audits were corrected. The requirements of this section were previously met and are documented in IIR 05000391/2009602, Attachment 2. Additional inspections beyond that IIR were performed under this inspection attribute and are documented in the following inspection report:

- 05000391/2012604 Section C.1.8 (ADAMS Accession No. ML12167A212)

During this inspection period, the inspectors observed the RPV lower internal lifts and records for the final construction related placement and protection of the lower internals.

The following samples were inspected:

- IP 50055 Section 02.02.b - one sample

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusion

The Reactor Vessel and Internals Record Review (IP 50055) previously met all the requirements as outlined in IIR 05000391/2009602, Attachment 2. The long dormancy and newly established storage/protection conditions of the RPV and internals required additional inspections under this procedure. The inspectors observed final construction related placement of the lower internals as well as the final construction related records documenting the safety-related work. As a result, IP 50055 is considered closed; however, additional inspections may be performed to include the installation of the upper internals at the NRC's discretion.

C.1.4 Preservice Inspections – Observation of Work and Work Activities (Inspection Procedure 73053)

a. Inspection Scope

Background: This IP provided guidance to the inspectors in reviewing the preservice inspection (PSI) program to ascertain whether the onsite preservice inspection of Class 1, 2, and 3 pressure retaining components are performed in accordance with regulatory requirements and applicant commitments.

Inspection Activities: The inspectors observed personnel performing direct visual examinations of core support structures, specifically the upper control rod drive mechanisms (CRDM) seismic restraints listed in American Society of Mechanical Engineers (ASME) Section XI. The NRC inspectors noted that the examiners performed the task in accordance with Appendix B, VT-3 Visual Examination, of the applicant's procedure, N-VT-1, "Visual Examination Procedure for ASME Section XI Preservice and Inservice," Rev. 46. The inspectors observed the personnel inspecting for proper clearances, potential damage, loose or missing parts, debris, wear and corrosion, and integrity of welded connections.

In addition, the inspectors reviewed inspection results, photographic images, and video recordings of the reactor pressure vessel lower internal core barrel support structure B-N-3 (2) as described in the work procedure WBN-2 PSI, "PSI Program Plan," Rev. 9. The inspections were conducted to verify the visual examinations were performed and recorded in accordance with ASME Section XI.

The inspectors also reviewed the qualification records for the personnel performing the visual examinations. The inspectors noted that the records included employer's name; person certified; activity qualified to perform; level of qualification; effective period of certification; signature of employer's designated representative; basis used for certification; annual visual acuity and color vision examination; and periodic recertification.

b. Observations and Findings

No findings were identified.

c. Conclusions

Work associated with preservice inspections listed above were completed in accordance with the approved procedures and met the requirements of ASME Section XI.

C.1.5 (Closed) Containment Penetrations (Mechanical) - Work Observation (Inspection Procedure 53053)

a. Inspection Scope

Background: The purpose of IP 53053 is to confirm adequate installation of mechanical containment penetrations. This IP covers material certification, method of assembly, protection, installation activities, nondestructive examination, and inspection as well as a semi-annual requirement to observe installation and confirm adequacy of documents used for installation. All containment penetrations were previously installed prior to initial suspension of the Unit 2 construction effort. A reconstitution of historic Unit 2 inspection results identified that inspection requirements were not met and that only minimal samples were noted. Because all containment penetrations had already been installed, inspections of work observation cannot be performed. To satisfy the requirements of this IP, it was recommended that inspection efforts focus on visual examination, records reviews, and witnessing local leak rate testing and/or integrated leak rate testing.

Previous inspections of containment penetrations have been documented in the following reports:

- IIR 05000391/2008007, Section E.1.1 (ADAMS Accession No. ML082110474)
- IIR 05000391/2008010, Section C.1.3 (ADAMS Accession No. ML090291033)
- IIR 05000391/2011604, Section C.1.8 (ADAMS Accession No. ML111810890)
- IIR 05000391/2013603, Section C.1.6 (ADAMS Accession No. ML13134A239)
- IIR 05000391/2013605, Section C.1.7 (ADAMS Accession No. ML13220A640)
- IIR 05000391/2015605, Section C.1.4 (ADAMS Accession No. ML15226A345)

Inspection Activity: The inspectors observed local leak rate testing of four mechanical/electrical penetrations performed under WO 116700704. The inspectors reviewed the work instructions, mock-up testing, and in-field testing setup against the applicable requirements, standards, and approved procedures to ensure they were met. The inspectors also reviewed measurement and test equipment (M&TE) records for equipment used for testing and verified that appropriate hold points were adhered to.

The following mechanical/electrical penetrations were inspected:

- WBN-2-PENT-304-0121E, Penetration for Electrical 2-PENT-293-0001
- WBN-2-PENT-304-0122E, Penetration for Electrical 2-PENT-293-0002
- WBN-2-PENT-304-0123E, Penetration for Electrical 2-PENT-293-0003
- WBN-2-PENT-304-0124E, Penetration for Electrical 2-PENT-293-0004

b. Observations and Findings

No findings were identified.

c. Conclusions

Work associated with containment penetrations listed above were completed in accordance with the approved procedures. Based on the activities reviewed in this and previous inspections, the inspectors concluded that work performance, work in progress, and completed work were being accomplished in accordance with NRC requirements, and applicant procedures. IP 53053 is considered closed; however, additional activities may be performed at the NRC's discretion.

C.1.6 Mechanical Components – Work Observation and Construction Refurbishment Process (Inspection Procedures 50073 and 37002)

a. Inspection Scope

Background: Problem evaluation report (PER) 172712 was initiated on June 1, 2009, to track the completion of NRC commitment NCO860025001, "Clean, flush, and inspect steam generators per QCT 4.3.6." This commitment was a result of Construction Deficiency Report (CDR) 391/83-16, Loose Material in Steam Generator. Commitment 114119980 was created on November 14, 2012, to track the commitment documented in PER 172712. In June of 2014, Westinghouse Electric Company issued the Watts Bar Unit 2 Steam Generator (S/G) Readiness Report. This report identified the need to perform the foreign object search and retrieval (FOSAR) inspection after hot functional testing as required by WO 112681952, in addition to the FOSAR inspections that have already been completed.

Inspection Activities: The inspector reviewed WO 112681952 and procedure 2-MI-3-3.015, "Steam Generator Secondary Side Maintenance Activities," Rev. 1. Additionally, the inspector observed parts of the FOSAR activities on S/G 2. This observation included the visual inspection of SG 2 in the tube-sheet, and tube lane areas, along with the hot and cold legs of the annulus. Following the completion of FOSAR activities, the inspector reviewed the Watts Bar U2R19 S/G Secondary Side Tubesheet FOSAR Report submitted by Vistas Corporation.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors did not identify any issues or concerns regarding the performance or quality of the FOSAR inspections of the Watts Bar Unit 2 S/Gs.

C.1.7 (Closed) Inspection Procedure 71111.07, Heat Sink Performance, Subissue: Essential Raw Cooling Water Testing; and Supplemental Safety Evaluation Report, Appendix HH, Open Item 90: Essential Raw Cooling Water Pumps Meeting General Design Criteria 5 Requirements for Two-Unit Operation (Inspection Procedures 92701 and 71111.07)

a. Inspection Scope

Background: This inspection utilized the requirements of IP 71111.07, Section 02.02.d.5(b), to provide reasonable assurance that the essential raw cooling water (ERCW) system flow balance testing could support dual unit operation as outlined in 10 *Code of Federal Regulations* (CFR), Part 50, Appendix "A", General Design Criterion (GDC) 5, "Sharing of structures, systems, and components." Inspection of these items were previously documented in IIRs 05000391/2014604 (ADAMS Accession No. ML 14177A214) and 05000391/2015603 (ADAMS Accession No. ML 15124A921).

Inspection Activities: Due to the overlapping relationship between these two items, the inspectors performed the following activities in order to address them collectively.

The inspectors reviewed the results of the Watts Bar Unit 2 flow balance test results associated with the "A" train and the "B" train of the ERCW system. The inspectors reviewed the plant response to the GDC-5 requirements and design basis flow requirements for the ERCW system, with the condition of one unit in accident conditions and the other unit in process of an orderly shutdown. The flow balance test requirements were derived by the applicant, based on several plant configurations, in order to demonstrate that the most limiting flow demand condition on ERCW with both units sharing ERCW output flows could be satisfied. The actual flow balance test results provided by the applicant were compared to the target flow rate requirements established by the test scoping document.

Test Deficiency Notices (TDNs) initiated for some target flow shortfalls were also reviewed to provide reasonable assurance that TDNs were dispositioned in accordance with applicant procedures.

Aside from the planned dual-unit flow tests, the applicant identified that the ERCW flow demand requirements during dual unit operation, based on the original two-pump ERCW design basis configuration testing, indicated that an additional (third) ERCW pump needed to be started in order to achieve the required flows during some design basis scenarios with a loss of coolant accident (LOCA) on one unit while the other unit in the process of completing a shutdown. For this case, additional regulatory commitments and procedural guidance was implemented to meet the intent of GDC-5 for shared systems between Unit 1 and Unit 2. Additional information on the three-pump ERCW test result review, and NRC observations during three-pump testing can be found in Section P.1.5 of this report.

The ERCW system discharge piping downstream of various heat exchangers serves as the safety-related source of water for the auxiliary feedwater (AFW) system in some Final Safety Analysis Report (FSAR) Chapter 15 accident analysis cases. Therefore, the inspectors reviewed a sample of heat exchanger discharge piping calculations and compared them to the FSAR Chapter 15 input assumption for AFW maximum input temperature of 120 degrees Fahrenheit (°F). This was done to verify that adequate net positive suction head (NPSH) was provided by the ERCW discharge conditions in order to satisfy the NPSH required at the AFW pump suction point. AFW pump NPSH calculations are also being updated in order to reflect additional testing conducted for a three-pump test configuration.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified. To resolve TDNs the applicant performed additional calculations or revised existing calculations to demonstrate that the achieved flows were acceptable. The inspectors identified that several calculations used in ERCW piping and heat exchanger discharge temperature evaluations exceeded the 120°F temperature assumed for the AFW inputs used in the FSAR Chapter 15 accident analyses. This discrepancy was entered into the corrective action program by CR 1072954 and is being evaluated by the applicant for impact.

c. Conclusions

The activities and reviews completed relative to the ERCW flow balance testing provide reasonable assurance that the required ERCW target flows were obtained during testing, TDNs evaluated shortfalls in target flow rates, and/or calculations could be revised to show that heat exchanger duty loads could be reduced with a subsequent reduction in demand on the ERCW system. Additionally, Technical Specifications (TS) amendments have been submitted to the NRC to address cases where the shared ERCW demand flows required additional pumps for dual unit operation. These reviews along with anticipated completion of in-process calculations, three-pump testing results documentation, and TS amendment completion, provide reasonable assurance that these items can be closed. Therefore, IP 71111.07 Section 02.02.d.5(b) and Supplemental Safety Evaluation Report (SSER) Appendix HH Item 90 are considered closed.

C.1.8 (Closed) Testing Piping Support and Restraint Systems, and Bulletin 88-11: Pressurizer Surge Line Thermal Stratification (Inspection Procedure 70370)

a. Inspection Scope

Background: These items were previously inspected in IIR 05000391/2015605 (ADAMS Accession No. ML 15226A345) and IIR 05000391/2015607 (ADAMS Accession No. ML15273A452). These inspections identified the need to review finalized records of completed testing and correction of any identified deficiencies. The previous inspection efforts satisfied the applicable preoperational testing samples outlined in IP 70370. Specifically:

- IP 70370, Section 02.01 a-e,

- IP 70370, Section 02.02, and
- IP 70370, Section 02.03.

Inspection Activities: The inspectors reviewed records of the pipe support testing to verify that the applicant has evaluated all piping support testing, that the results of the testing were within the established acceptance criteria, and that any deficiencies have been placed in the corrective action program and/or already corrected. This satisfied the following:

- IP 70370, Section 02.06.

Additionally, as part of Bulletin 88-11, the inspectors reviewed the applicant's procedures to verify that a differential temperature limit was established and maintained during testing. The inspectors also reviewed the test data for the pressurizer surge line movements to verify that the data was within the expected range and met engineering evaluation. The inspectors also reviewed established operational controls in applicant procedure 2-SI-68-44, "RCS Temperature/Pressure Limits and Pressurizer Temperature Limits," Rev. 4 to ensure that adequate guidance was provided to minimize surge line stratification.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusion

The inspectors determined that the applicant adequately inspected the condition of and measured the movement of supports on safety-related systems. IP 70370 will remain open until the operational requirements of sections 02.01 f-g, 02.04, and 02.05 are inspected. All preoperational section requirements have been completed. The inspectors determined that the applicant also adequately met the conditions required by Bulletin 88-11 for the pressurizer surge line thermal stratification. Bulletin 88-11 is considered closed.

P.1 Preoperational Activities

P.1.1 Preoperational Test Program Implementation Verification (Inspection Procedure 71302)

a. Inspection Scope

02.01 (Weekly Inspection Activities): The inspectors verified that the applicant's management control system was effectively discharging its responsibilities over the preoperational testing program by facility record review, direct observation of activities, tours of the facility, interviews, and discussions with applicant personnel. Preoperational testing activities during the inspection period included the following systems or portions thereof:

- System 074 – Residual Heat Removal

- System 070 – Component Cooling
- System 067 – Emergency Raw Cooling Water
- System 099 – Reactor Protection

As systems became available for preoperational testing, inspectors toured the accessible areas of the facility to make an independent assessment of equipment conditions, plant conditions, security, and adherence to regulatory requirements. The inspectors also reviewed the following, as available and on a sampling basis, during the tours:

- general plant/equipment conditions;
- plant areas for fire hazards - examined fire alarms, extinguishing equipment, actuating controls, firefighting equipment, and emergency equipment for operability and also verified that ignition sources and flammable material were being controlled in accordance with the applicant's procedures;
- activities in progress (e.g., maintenance, preoperational testing, etc.) were being conducted in accordance with the applicant's procedures;
- watched for abuse of installed instrumentation such as stepping or climbing on the instrumentation that could affect the calibration or ability to function;
- listened for the public address system announcements to determine that blind spots do not exist; (i.e., cannot be heard clearly enough to be understood);
- construction work force was authorized to perform activities on systems or equipment; and
- looked for uncontrolled openings in previously cleaned or flushed systems or components.

02.02 (Monthly Inspection Activities) During this inspection period, the inspectors reviewed the Unit 2 planned/completed surveillances to ensure required preventative maintenance was incorporated into a schedule for accomplishment.

02.02 (Quarterly Inspection Activities) The inspectors reviewed jurisdictional controls to verify that maintenance activities were performed by the proper group and sampled preventative maintenance activities to ensure satisfactory completion. The inspectors also witnessed testing and interviewed personnel to verify that the method for testing was current, that methods existed to assure personnel involved were knowledgeable of the test, that approved change methodologies were followed, that criteria for test interruptions were discussed, and that test deficiencies were properly documented.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

Introduction: The inspectors identified a Severity Level (SL) IV non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XII, "Control of Measuring and Test Equipment," for the failure to establish measures to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality were properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within

necessary limits. Specifically, the applicant failed to perform calibrations on a specified interval for digital stopwatches that were used for activities affecting quality.

Description: On August 15, 2015, the inspectors identified that the applicant was performing safety-related preoperational testing with stopwatches that were classified as one time calibration only. For each of the stopwatches used, the inspectors noted that the next calibration date was marked as “not applicable (N/A).” The inspectors noted that for two specific stopwatches, MT&E # 905260 and 904512, the initial calibration date was November 11, 1996 and August 5, 2006, respectively.

The applicant stated that the calibration process was in accordance with Section 3.2.6., Establishing Calibration Instructions and Calibration Schedules, of the TVA fleet procedure, NPG-SPP-06.4, “Measuring and Test Equipment,” Rev. 3, which stated in part, “that M&TE or plant standards which are not subject to drift or wear that could affect accuracy shall receive an initial certified calibration but are not required to be rechecked periodically. (Some examples may be manometers, bend radius gauge, digital stop watches, glass thermometers, and glass hydrometers.)”

The inspectors noted that in Appendix B, Regulatory Guide Conformance Status, of procedure TVA-NQA-PLN89-A, “Nuclear Quality Assurance Plan (NQAP) (Quality Assurance Program Description),” Rev. 29, the applicant committed to Section 5.2.16 of ANSI N18.7 – 1976/ANS-3.2, “Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants” and Section 13 of ANSI N45.2 – 1971/ANS-3.2, “Quality Assurance Program Requirements for Nuclear Power Plants.”

Section 5.2.16 of ANSI N18.7 – 1976/ANS-3.2, stated in part, that “tools, instruments, testing equipment and measuring devices used for measurements, tests and calibration shall be of the proper range and type and shall be controlled, calibrated and adjusted and maintained at specified intervals or prior to use to assure the necessary accuracy of calibrated devices. “Section 13 of ANSI N45.2 – 1971/ANS-3.2, stated in part, that “measures shall be established and documented to assure that tools, gages, instruments, and other inspection, measuring, and testing equipment and devices used in activities affecting quality are of the proper range, type, and accuracy to verify conformance to established requirements. To assure accuracy, inspection, measuring, and test equipment shall be controlled, calibrated, adjusted, and maintained at prescribed intervals or prior to use against certified equipment having known valid relationships to nationally recognized standards. If no national standards exist, the basis for calibration shall be documented. This requirement is not intended to imply a need for special calibration and control measures on rulers, tape measures, levels, and such other devices, if normal commercial practices provide adequate accuracy. The method and interval of calibration for each item shall be defined and shall be based on the type of equipment, stability characteristics, required accuracy, and other conditions affecting measurement control.”

The inspectors noted on the Central Laboratories Services (CLS) calibration report for stopwatch # 905260, it stated in part, that “This is to certify that all instrumentation, testing methods and personnel used comply with the requirements of the CLS Quality Assurance Program which is designed to meet the requirements of 10 CFR 50 Appendix B and ANSI N45.2-1971. Standards used are traceable to the National Institute of Standards and Technology (NIST), officially recognized agencies, commercially accepted practices or natural physical constants.”

The inspectors noted that NIST Special Publication 960-12, "Stopwatch and Timer Calibrations (2009 edition)", state in part, that "because of their slow aging rate, stopwatches tend to produce very repeatable results over long periods of time. When compared to the requirements for stopwatch calibrations discussed in Section 3 of the NIST, the small changes in frequency due to aging are usually insignificant. This makes it possible for a laboratory to allow long intervals (perhaps exceeding one year) between calibrations."

The inspectors determined that the applicant's failure to perform calibrations at a specified interval was a performance deficiency. This performance deficiency was considered to be more than minor because the finding represents an inadequate process, procedure, or quality oversight function, that if left uncorrected, could adversely affect the quality of the fabrication, construction, testing, analysis, or records of a safety-related SSC, in accordance with Appendix C, Minor Violations and Findings, of IMC 2517, "Watts Bar Unit 2 Construction Inspection Program," dated June 6, 2013. This issue was entered into the corrective action program as CR 1083002. The inspectors reviewed this finding against cross-cutting area components as described in IMC 0310, "Components Within the Cross-Cutting Areas," and determined that no cross-cutting aspect applied.

Enforcement: 10 CFR 50, Appendix B, Criterion XII, "Control of Measuring and Test Equipment," requires that measures shall be established to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits. Contrary to the above, since September 27, 2010, the applicant failed to establish measures to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality were properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits. Specifically, the applicant failed to perform calibrations on a specified interval for digital stopwatches that were used for activities affecting quality.

This finding was determined to be a SL IV violation using Section 6.5 of the Enforcement Policy. Because this was a SL IV violation and because it was entered into the corrective action program as CR 1083002, this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. This NCV is identified as NCV 05000391/2015608-01, "Failure to Perform Stopwatch Calibrations on a Specified Interval."

c. Conclusion

With the exception of the aforementioned finding, the applicant's implementation of the preoperational test program was in accordance with procedures for those activities observed during the inspection period.

P.1.2 (Closed) Comparison of As-Built Plant to FSAR Description (Inspection Procedure 37301)

a. Inspection Scope

Background: The purpose of IMC 2513, Light Water Reactor Inspection Program - Preoperational Testing and Operational Preparedness Phase, issue date January 1, 1984, is to verify through direct observation, personnel interviews, and review of facility records that:

- systems and components important to the safety of the plant are fully tested to demonstrate that they satisfy their design requirements;
- management controls and procedures, including quality assurance programs, necessary for operation of the facility have been documented and implemented.

IMC 2513 defines the inspection program that supports the issuance of an operating license. IMC 2513 requires, through IP 37301, that a sample of technical specification systems be inspected to verify that the as-built plant conforms to the commitments contained in the FSAR. The following inspection was performed in relation to satisfying the required comparison of the as-built plant to the FSAR description.

Inspection Activities: IP 37301 was previously inspected and documented in IIR 05000391/2015607, Section P.1.3 (ADAMS Accession No. ML15273A452). During this inspection, the inspectors selected the following safety systems to complete inspection samples as described in paragraph 03.01a of IP 37301:

- System 001, Main Steam
- System 015, Steam Generator Blowdown
- System 018, Diesel Generator Fuel Oil
- System 020, Diesel Lube Oil
- System 030F, Containment Vent Air Cleanup
- System 030I, Control Rod Drive Mechanism Coolers
- System 030J, Containment Purge
- System 030AB, Auxiliary Building Gas Treatment
- System 032, Control Air
- System 063, Safety Injection
- System 078, Spent Fuel Cooling
- System 090, Radiation Monitoring
- System 092, Neutron Monitoring
- System 094, Incore Instrumentation
- System 235, 120 AC Vital Power
- System 236, 120 DC Vital Power

Additionally, the inspectors selected the following safety systems to complete inspection samples as described in paragraph 03.01b of IP 37301:

- System 003B, Auxiliary Feedwater
- System 030E, Containment Air Return Fans
- System 046B, Auxiliary Feedwater Controls
- System 061, Ice Condenser
- System 062, Chemical and Volume Control

- System 074, Residual Heat Removal
- System 268, Permanent Hydrogen Mitigation
- System 026, High Pressure Fire Protection

The inspectors performed the following to verify that the as-built condition and technical specification function for each system conforms to the commitments in the FSAR:

- reviewed the latest revision of the mechanical configuration control drawing(s) to verify agreement with the current FSAR and system description;
- reviewed the latest revision of the control and logic instrumentation configuration control drawing(s) to verify agreement with the description of instrumentation and controls contained in the FSAR;
- for systems selected to accomplish paragraph 03.01a of IP 37301, performed a limited scope walkdown to verify that significant components, including control and logic instrumentation (where applicable), are as described in drawings, technical specifications, and the FSAR; and
- for systems selected to accomplish paragraph 03.01b of IP 37301, performed a full system walkdown, of at least one train, to verify that significant components, including control and logic instrumentation (where applicable), are as described in drawings, technical specifications, and the FSAR, and that the technical specification surveillances could be accomplished with the instrumentation as installed.

In instances where inspectors determined that the latest revision of the configuration control drawings was not in agreement with the current FSAR drawings or descriptions, the inspectors:

- verified that the design change was reviewed, processed, and implemented in accordance with procedural controls,
- verified that the cognizant test engineer is aware of the discrepancy and that the change will not invalidate a test procedure or test result, and
- verified that any appropriate proposed changes to update the FSAR and technical specifications have been submitted to or is being processed for submittal to NRR.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

Introduction: The inspectors identified a SL IV NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The inspectors identified two examples where the applicant failed to prescribe documented instructions that were appropriate to the circumstance for testing to demonstrate operability of safety-related structures, systems, and components (SSCs). Specifically, performance of a surveillance test instruction resulted in unacceptable preconditioning of the containment air return (CAR) backdraft damper prior to recording the force required to open the backdraft damper.

Additionally, surveillance test instructions resulted in unacceptable preconditioning of the hydrogen igniter circuits prior to recording the required current or temperature output.

Description: On September 17, 2015, the inspectors performed a review of surveillance instruction 2-SI-30-26-A. This surveillance instruction is required in order to meet the TS Surveillance Requirements (SRs) associated with the containment air return system. The inspectors identified an example of preconditioning on backdraft damper 2-BKD-30-550 associated with CAR fan 2A-A prior to verifying TS SR 3.6.10.3. Step 6.0[36] of 2-SI-30-26-A records the force required to open the backdraft damper, which is used to calculate the required torque as specified in the test acceptance criteria. The inspectors identified that an identical step is performed earlier in the procedure at step 6.0[11], but this recorded force is not used to verify the acceptance criteria. Additionally, the damper is cycled open and closed at step 6.0[12].

TVA's Nuclear Quality Assurance Program states, in part, that "controls shall be established to ensure that required testing is identified and performed in accordance with procedures, which incorporate engineering requirements." NPG-SPP-06.9.2, Rev. 4, "Surveillance Test Program" incorporates the definition of preconditioning per the NRC Inspection Manual Part 9900:

"The alteration, variation, manipulation, or adjustment of the physical condition of an SSC before Technical Specification surveillance or ASME code Testing"

Based on the applicant's surveillance instruction test procedure, the inspectors determined that the applicant failed to meet the NPG-SPP-06.9.2 Rev. 4, "Surveillance Test Program," procedure and preconditioned the backdraft damper 2-BKD-30-550.

The inspectors identified a second example of unacceptable preconditioning on September 22, 2015. The inspectors performed a review of surveillance instructions 2-SI-268-1 and 2-SI-268-2 for both trains of the hydrogen mitigation system (HMS). These surveillance instructions are required to demonstrate TS operability of the HMS. The inspectors identified an example of preconditioning on the hydrogen igniter circuits prior to verifying TS SR 3.6.8.1, TS SR 3.6.8.2, and TS SR 3.6.8.3. Specifically, the load side voltage for the hydrogen igniter circuits are adjusted to within a certain voltage band prior to recording the as-found current readings for each hydrogen igniter. Adjusting the voltage on the system has a direct impact on the current readings for each igniter. The practice of adjusting the load side voltage prior to documenting the as-found current readings for each igniter is an example of unacceptable preconditioning.

The inspectors determined that the applicant's failure to establish adequate surveillance test instructions, that caused unacceptable preconditioning of SSCs, was a performance deficiency. The inspectors determined that the performance deficiency was more than minor because it represented an inadequate process or procedure that if left uncorrected, could adversely affect testing of safety-related SSCs, in accordance with Appendix C, Minor Violations and Findings, of IMC 2517, "Watts Bar Unit 2 Construction Inspection Program," dated June 6, 2013. In one example, this condition could give a false "as-left" torque value of the CAR backdraft damper, as required by the TS surveillance acceptance criteria. For the hydrogen igniter circuits, this condition could give a false "as-found" amperage value in the circuits and could affect the measured temperature of the hydrogen igniter elements, as required by the TS surveillance

acceptance criteria. Performing the surveillances as written could mask the conditions used to determine equipment operability. These issues were entered into the corrective action program as CR 1083315 and CR 1088279. The inspectors reviewed this finding against cross-cutting area components as described in IMC 0310, "Components Within the Cross-Cutting Areas," and determined that no cross-cutting aspect applied.

Enforcement: 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, 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."

NPG-SPP-06.9.2, "Surveillance Test Program," incorporates the definition of preconditioning per the NRC Inspection Manual Part 9900:

"The alteration, variation, manipulation, or adjustment of the physical condition of an SSC before Technical Specification surveillance or ASME Code Testing".

NPG-SPP-06.9.2 states that "Surveillance sequences and alignment activities that are implemented prior to a surveillance test should be intended to facilitate testing or prevent unnecessary wear and damage to equipment and/or systems. Test procedure steps that exercise equipment prior to testing activities are not allowed unless they provide a specific benefit, such as improved safety, reduced wear and tear, or facilitate better test methods."

Contrary to the above, on September 17, 2015 and September 22, 2015, the inspectors identified test procedure steps that exercise equipment prior to surveillance testing activities, without a specific benefit. Specifically, surveillance instruction 2-SI-30-26-A contains steps that result in unacceptable preconditioning of the backdraft damper associated with the containment air return system. Surveillance instructions 2-SI-268-1 and 2-SI-268-2 contain steps that result in unacceptable preconditioning of the HMS igniter circuit associated with the HMS.

This finding was determined to be a SL IV violation using Section 6.5 of the Enforcement Policy. Because this was a SL IV violation and because it was entered into the corrective action program as CR 1083315 and CR 1088279, this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. This NCV is identified as NCV 05000391/2015608-02, "Preconditioning Identified in Surveillance Instructions."

c. Conclusions

The inspectors determined that the as-built condition of the technical specification portions of the inspected systems conform to the commitments contained in the FSAR. Based on the above activities and the results of previous inspection, the inspection requirements of IP 37301 have been completed. IP 37301 is closed.

P.1.3 Preoperational Test Procedure Review (Inspection Procedures 63050 and 70307)

a. Inspection Scope

Background: The purpose of IMC 2512, Light Water Reactor (LWR) Inspection Program – Construction Phase, issue date December 17, 1986, is to provide inspection requirements and policy for implementation of the inspection program during construction and major plant modifications. Appendix I of 2512 defines the current IPs that are applicable to construction and major modification activities, along with condensed scheduling information, which includes IP 63050, Containment Structural Integrity Test, issue date July 1, 1983. The purpose of this IP is to determine whether the containment structural integrity test (SIT) program, instructions, procedures, actual test performance, and evaluation of test results are consistent with regulatory requirements and applicant commitments. The containment SIT demonstrates the capability of the primary reactor containment to withstand specified internal pressure loads.

The purpose of IMC 2513, LWR Inspection Program - Preoperational Testing and Operational Preparedness Phase, issue date January 1, 1984, is to verify through direct observation, personnel interviews, and review of facility records that systems and components important to the safety of the plant are fully tested to demonstrate that they satisfy their design requirements, and management controls and procedures, including quality assurance programs, necessary for operation of the facility have been documented and implemented. Appendix A, LWR – Preoperational Testing Phase, of IMC 2513 describes the inspection program to verify systems and components important to safety of the plant are fully tested to demonstrate that they satisfy their design requirements. IP 70307, Containment Integrated Leak Rate Test Procedure Review, issue date March 7, 1986, is an IMC 2513, Appendix A, inspection procedure. The purpose of this IP is to ascertain if the applicant's procedure for the containment integrated leak rate tests (CILRTs) complies with regulatory requirements, guidance, and applicant commitments, and to evaluate the technical adequacy of this procedure to determine containment leak tight integrity.

The following inspection was performed to satisfy the required procedural review of the SIT and CILRT. Because of the consecutive performance of the two tests, the applicant developed a hierarchy of procedures and WOs that controlled similar prerequisites for the two tests, along with the series of actions for both containment pressurization and depressurization. The applicant also had specific procedures and WOs to perform the individual requirements for each of the containment tests. The inspectors reviewed governing instructions, which included TVA programmatic, operations, and maintenance procedures, and vendor subject matter expert (SME) procedures and WOs, to verify the technical adequacy of not only the documents themselves but how the documents interfaced with each other for both tests.

Inspection Activities: The inspectors performed Section 02.01; Review of SIT Program, Procedures, and Instructions; of IP 63050 and IP 70307 in its entirety. The inspectors performed interviews, walkdowns, and documentation reviews of the SIT/CILRT programs, procedures, WO instructions, and essential drawings, to verify that the instructions given to perform the SIT and CILRT met the applicable requirements. Applicable NRC requirements included: NUREG-0800, "Standard Review Plan," Rev. 3; Regulatory Guide 1.68; and the Safety Evaluation Report (SER). Applicable code requirements included: commitments for American National Standards Institute (ANSI), American Nuclear Society (ANS), and ASME. Commitments were also outlined in the FSAR and proposed TSs. The inspectors specifically reviewed the following:

- 2-PTI-064-02, "Containment Integrated Leak Rate Test (CILRT)," Rev. 1;
- 2-TSD-064-02, "WBN2 CILRT – Test Scoping Document," Rev. 2;
- WO 116347436, "Perform 2-PTI-64-02;"
- NEDP-14, "Containment Leak Rate Programs," Rev. 11;
- 2-TI-360.01, "WBN Containment Leak Rate Programs," Rev. 1;
- 2-SI-0-703, "Containment Integrated Leak Rate Test," Rev. 2, one time only (OTO)-1, and OTO-2;
- WO 116348205, "Perform 2-SI-0-703;"
- 2-TI-120, "CILRT Support Functions," Revs. 2, OTO-1, and OTO-2;
- WO 116585126, "Perform 2-TI-120;"
- 25402-000-GPP-0000-N3506, "Pressure Testing of Piping, Tubing, and Components (Bechtel)," Rev. 13;
- WO 115689350, "Structural Integrity Test;" and
- 25402-011-V1b-NEE0-00002-001, "Containment Vessel Structural Integrity Test Procedure (Graftel)," dated May 5, 2015.

In addition, the inspectors reviewed procedures and WOs to verify the following were included for both the SIT and CILRT:

- the containment structure was closed out and a general inspection of the accessible portions of interior and exterior containment surfaces was performed in accordance with applicant's procedures;
- the test pressure and hold time used, maximum rate of pressurization, including pressurization increments, and depressurization of the containment were specified;
- logging of required test parameters, leakage rates, and observations, which included atmospheric and containment pressure, temperature, dew point temperature, and liquid level (e.g., suppression pool or pressurizer level) was on a pre-determined frequency basis;
- proper system alignment and configuration control for testing was listed. The inspectors reviewed recommended lineups for systems penetrating the primary containment versus actual system configurations (i.e. tag-outs, corrective maintenance, barrier impairments); and
- determination of satisfactory test results was documented, which included the calculation/design basis for the acceptance criteria and how instrument uncertainty was accounted for.

The inspectors reviewed SIT procedures and WOs to specifically verify the following:

- Current calibration records existed for the two pressure gauges used and the temperature sensors used to verify shell temperature during the SIT. The inspectors verified the type, location, range, accuracy, and calibration of these instruments.
- The hold time for the SIT pressure was to be at least ten minutes. The inspectors verified that after the test pressure had been attained, the procedure or WO directed that the pressure was to be reduced to the specified fraction of the test pressure or design pressure and directed the examination of leakage for all required joints and penetration units at locations specified by the procedure and/or ASME code.

The inspectors noted that in 1978, an overload pressure test, i.e. SIT, was performed by TVA and Chicago Bridge and Iron (CBI) for the Watts Bar Unit 2 containment; in which the seam welds in the vessel shell and dome were solution film tested. In addition, in 1975 during original construction, the steel vessel structural containment welds were tested using vacuum box testing and radiography. As a result, the applicant generated CR 874803 and concluded that additional solution film testing during the 2015 SIT was not required for welds previously tested. In addition, the applicant concluded that the SIT being performed in 2015 was to verify the leak tightness of all the steel containment vessel welds added since the testing of the vessel in 1978. The inspectors verified the CR and corrective action documentation in accordance with ASME Boiler and Pressure Vessel Code, Section III and the Authorized Nuclear Inspector (ANI) program. Specifically, the welds that were tested and accepted by TVA and CBI from the 1978 testing were:

- welds in the shell and dome;
- welds of penetration sleeves to the containment vessel;
- insert to shell welds;
- assessable seams in equipment access hatch and cover; and
- for personnel locks, bulkheads, and portions of barrel between interior and exterior doors.

The inspectors reviewed CILRT procedures and WOs to specifically verify the following:

- the test was in compliance with 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," and ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements," which included Type A test requirements; local leak rate testing (LLRT) requirements; instrument calibrations and pretest checks; containment pressurization, depressurization, stabilization, and hold times; verification test; and termination limits;
- the test instruments used were within calibration at the initiation of the CILRT. The inspectors reviewed calibration records for temperature, relative humidity, and pressure sensors used in the CILRT to verify that all equipment was properly calibrated to the range and accuracy required by the applicable code;
- the containment area survey for temperature differentials and humidity had been performed and the location of the sensors was determined based on this survey and placed in locations representative of their assigned subvolumes; and
- the applicant's calculations to determine the Unit 2 containment volume and the maximum allowable leakage rates was up-to-date and accurately reflected equipment and system modifications in containment that could affect the calculations.

The inspectors reviewed FSAR table 6.2.1-13 to derive the total containment volume and verified that the applicant's calculations included the new volumes for the ice condenser, the updated steam generator sizing, and the models that were provided by Westinghouse for the unit. The inspectors also reviewed the material list of items left in the unit that would have an impact on total volume. The inspectors performed walkdowns of containment to verify the accuracy of the estimation of material that would be left inside the containment building during testing.

Additionally, the inspectors reviewed the preoperational test procedures to verify that the procedures contained the following administrative good practice attributes:

- the title described the purpose of the procedure;
- the cover page had appropriate information and approval signatures;
- each page had appropriate identification information;
- the last page was clearly identifiable by markings;
- the procedure purpose/objectives; initial conditions, prerequisites, precautions, required tools, reference documents, and coordination requirements was clearly listed;
- signoff requirements including concurrent and independent verification steps were established, where appropriate;
- equipment alignment and restoration instructions were clear and concise;
- equipment identifiers were accurate;
- instrumentation units were consistent for data collection;
- graphs, charts, tables, data sheets, and work sheets were clearly usable;
- calculation sheets were technically accurate;
- clear instructions existed for test activities involving multiple test personnel;
- clear concise testing steps existed with quantitative acceptance criteria listed; and
- overall sequence of the procedure was consistent with obtaining the intended results.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors determined that the applicant's test procedures and WOs for the SIT and CILRT were technically adequate and were written in a manner consistent with the guidance of procedure SMP-8.0, "Watts Bar Nuclear Plant Unit 2 Administration of Preoperational Test Instructions," Rev. 11. This completes the SIT and CILRT procedure review of preoperational test procedures, including 2-PTI-064-01. With respect to IP 70307, this IP was performed in its entirety. However, only Section 02.01 of IP 63050 was performed during this inspection. The inspection of IP 63050, Section 02.02, "SIT Witness," is covered in Section P.1.8 of this report. Further inspection of Section 02.03, "Review of Test Records," is required for IP 63050.

P.1.4 Preoperational Test Witnessing (Inspection Procedure 70312)

a. Inspection Scope

Background: The inspectors witnessed portions of the data storm testing on the digital control system (DCS). The DCS is a nonsafety-related digital control system used to control most of the balance of plant equipment. The applicant conducted a test on the DCS system to demonstrate its continued reliability when subjected to large amounts of digital input. The test was performed under WO 117056933 and began on October 1,

2015. The data storm was performed on each of four network switches located in both the auxiliary instrument room and the auxiliary control room. The test subjected the equipment to one broadcast and one multicast storm to simulate the worst case component failure on the DCS network. The objective of the test was to demonstrate that even during a network data storm the control processors and the controls remain functional.

Inspection Activities: The inspectors witnessed the worst case data storm testing on one of the four switches to verify that the control processors and the controls remained functional.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified. During the data storm conditions there was some expected loss of redundancy and monitoring capability as anticipated, however the controls all remained functional as demonstrated by operator manipulation during the test.

c. Conclusion

The inspectors concluded that the applicant's data storm test was performed successfully and met its objectives.

P.1.5 Preoperational Test Witnessing (Inspection Procedure 70312)

a. Inspection Scope

Background: The purpose of preoperational test inspection is to verify through direct observation, personnel interviews, and review of facility records that:

- systems and components important to the safety of the plant are fully tested to demonstrate that they satisfy their design requirements; and
- management controls and procedures, including QA programs, necessary for operation of the facility have been documented and implemented.

IMC 2513 defines the inspection program that supports the issuance of an Operating License. IMC 2513 requires the preoperational test witnessing of the mandatory tests defined in IMC 2513 and five of the primal tests defined in IMC 2513. The following inspection was performed in relation to satisfying the required preoperational test witnessing.

Inspection Activities: The inspectors witnessed activities associated with the performance of preoperational test instruction 2-PTI-067-04, "Essential Raw Cooling Water (ERCW) Dual Unit Shutdown Flow Test," Rev. 0, and WO 116923934, "Collect CCS Train B Flow Data With Both CCS Pumps In Service," to verify that the testing was conducted in accordance with approved procedures and to verify the adequacy of test program records and preliminary evaluation of test results. The following component tests were selected for inspection:

- Section 6.2, ERCW Train B Dual Unit Shutdown Flow Test, and

- WO 116923934, Collect CCS Train B Flow Data With Both CCS Pumps In Service.

The inspectors assessed the following attributes associated with this test observation:

- all test personnel were on station and had the latest revision of the procedure;
- test prerequisites were performed;
- plant systems were in service to support the test;
- test equipment was installed and within calibration;
- testing was performed in accordance with the approved procedure;
- test interruptions and continuations were handled in accordance with approved procedures and documented in the chronological test log;
- testing events and discrepancies were properly documented in the test deficiency log;
- testing was executed and coordinated properly;
- data was properly collected;
- temporary equipment was installed and tracked appropriately;
- administrative test controls were properly followed; and
- test personnel were using approved drawings and vendor manuals.

The inspectors observed the tests to verify that the overall test acceptance was met. The inspectors conducted a review with the responsible test engineer to assure that the preliminary test evaluations were consistent with the inspector's observations. During the tests, the inspectors observed important data gathering activities to ensure the data was properly gathered and recorded. A post-test cursory review of the test data was performed to verify legibility, traceability, and permanence of the data sheet entries.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors determined that the applicant's test procedure was performed in a manner consistent with the guidance of procedure SMP-9, Watts Bar Nuclear Plant Unit 2, "Conduct of Test," Rev. 6.

P.1.6 Preoperational Test Witnessing (Inspection Procedures 70312 and 70348)

a. Inspection Scope

Background: The background for this preoperational test witnessing is the same as that in the background section of P.1.5.

Inspection Activities: The inspectors witnessed activities associated with the performance of preoperational test instruction 2-PTI-003A-01, "Feedwater Isolation Valves," Rev. 0 to verify that the testing was conducted in accordance with approved procedures and to verify the adequacy of test program records and preliminary

evaluation of test results. The following component tests were selected for inspection of this item:

- Section 6.1, Steam Generator Loop 1 Valves
- Section 6.2, Steam Generator Loop 2 Valves
- Section 6.4, Steam Generator Loop 4 Valves
- Section 6.5.1, 2-FCV-3-250: Steam Generator 1 Feedwater Regulating Isolation Valve
- Section 6.5.4, 2-FCV-3-253: Steam Generator 3 Feedwater Regulating Isolation Valve

The inspectors assessed the following attributes associated with this test observation:

- all test personnel were on station and had the latest revision of the procedure;
- test prerequisites were performed;
- plant systems were in service to support the test;
- test equipment was installed and within calibration;
- testing was performed in accordance with the approved procedure;
- test interruptions and continuations were handled in accordance with approved procedures and documented in the chronological test log;
- testing events and discrepancies were properly documented in the test deficiency log;
- testing was executed and coordinated properly;
- data was properly collected;
- temporary equipment was installed and tracked appropriately;
- administrative test controls were properly followed; and
- test personnel were using approved drawings and vendor manuals.

The inspectors observed the tests to verify that the overall test acceptance was met. The inspectors conducted a review with the responsible test engineer to assure that the preliminary test evaluations were consistent with the inspector's observations. During the tests, the inspectors observed important data gathering activities to ensure the data was properly gathered and recorded. A post-test cursory review of the test data was performed to verify legibility, traceability, and permanence of the data sheet entries.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors determined that the applicant's test procedure was performed in a manner consistent with the guidance of procedure SMP-9, "Watts Bar Nuclear Plant Unit 2, Conduct of Test," Rev. 6. This concludes the planned witnessing of 2-PTI-003A-01; however, additional inspection may be performed, at the discretion of the NRC, should additional testing occur.

P.1.7 Preoperational Test Witnessing (Inspection Procedures 70312 and 70436)

a. Inspection Scope

Background: The background for this preoperational test witnessing is the same as that in the background section of P.1.5.

Inspection Activities: The inspectors witnessed activities associated with the performance of preoperational test instruction 2-PTI-074-01, "Residual Heat Removal System Pump/Valve Logic," Rev. 1 to verify that the testing was conducted in accordance with approved procedures and to verify the adequacy of test program records and preliminary evaluation of test results. The following component tests were selected for inspection of this item:

- Section 6.3, Valve 2-FCV-74-1 Logic Test
- Section 6.4, Valve 2-FCV-74-2 Logic Test
- Section 6.5, Valve 2-FCV-74-8 Logic Test
- Section 6.6, Valve 2-FCV-74-9 Logic Test

The inspectors assessed the following attributes associated with this test observation:

- all test personnel were on station and had the latest revision of the procedure;
- test prerequisites were performed;
- plant systems were in service to support the test;
- test equipment was installed and within calibration;
- testing was performed in accordance with the approved procedure;
- test interruptions and continuations were handled in accordance with approved procedures and documented in the chronological test log;
- testing events and discrepancies were properly documented in the test deficiency log;
- testing was executed and coordinated properly;
- data was properly collected;
- temporary equipment was installed and tracked appropriately;
- administrative test controls were properly followed; and
- test personnel were using approved drawings and vendor manuals.

The inspectors observed the tests to verify that the overall test acceptance was met. The inspectors conducted a review with the responsible test engineer to assure that the preliminary test evaluations were consistent with the inspector's observations. During the tests, the inspectors observed important data gathering activities to ensure the data was properly gathered and recorded. A post-test cursory review of the test data was performed to verify legibility, traceability, and permanence of the data sheet entries.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors determined that the applicant's test procedure was performed in a manner consistent with the guidance of procedure SMP-9, "Watts Bar Nuclear Plant Unit 2, Conduct of Test," Rev. 6. This concludes the planned witnessing of 2-PTI-074-01;

however, additional inspection may be performed, at the discretion of the NRC, should additional testing occur.

P.1.8 Preoperational Test Witnessing (Inspection Procedures 63050 and 70313)

a. Inspection Scope

Background: As previously documented in Section P.1.3, the purpose of IMC 2512 is to provide inspection requirements and policy for implementation of the inspection program during construction and major plant modifications. Appendix I of 2512 includes IP 63050, "Containment Structural Integrity Test," issue date July 1, 1983. The purpose of this IP is to determine whether the containment structural integrity test (SIT) program, instructions, procedures, actual test performance, and evaluation of test results are consistent with regulatory requirements and applicant commitments. The containment SIT demonstrates the capability of the primary reactor containment to withstand specified internal pressure loads.

The purpose of IMC 2513 is to verify through direct observation, personnel interviews, and review of facility records that systems and components important to the safety of the plant are fully tested to demonstrate that they satisfy their design requirements, and management controls and procedures, including quality assurance programs, necessary for operation of the facility have been documented and implemented. Appendix A of IMC 2513 includes IP 70313, Containment Integrated Leak Rate Test, issue date October 3, 1985. The purpose of this IP is to ascertain through inspector observation, records review, and independent calculations whether the CILRT is being properly conducted, and to independently verify the acceptability of test results.

The following inspection was performed to satisfy the required test witnessing of the SIT and CILRT. Because of the consecutive performance of the two tests, the applicant developed a hierarchy of procedures and WOs that controlled similar prerequisites for the two tests, along with the series of actions for both containment pressurization and depressurization. The applicant also had specific procedures and WOs to perform the individual requirements for each of the containment tests.

Inspection Activities: The inspectors performed section 02.02, SIT Witness, of IP 63050 and IP 70313 in its entirety. The inspectors performed interviews, walkdowns, direct testing observations, and documentation reviews of the SIT/CILRT procedures, WO instructions, and essential drawings, to verify that the testing performed for the SIT and CILRT met the applicable requirements. Applicable NRC requirements included: NUREG-0800, "Standard Review Plan," Rev. 3; Regulatory Guide 1.68; and the SER. Applicable code requirements included: commitments for ANSI, ASME, and ASME. Commitments were also outlined in the FSAR and proposed Technical Specifications. The inspectors specifically observed the following being performed:

- 2-PTI-064-02, "Containment Integrated Leak Rate Test (CILRT)," Rev. 1;
- WO 116347436, "Perform 2-PTI-64-02;"
- 2-SI-0-703, "Containment Integrated Leak Rate Test," Rev. 2, OTO-1 and OTO-2;
- WO 116348205, "Perform 2-SI-0-703;"
- 2-TI-120, "CILRT Support Functions," Rev. 2, OTO-1 and OTO-2;

- WO 116585126, "Perform 2-TI-120;"
- 25402-000-GPP-0000-N3506, "Pressure Testing of Piping, Tubing, and Components (Bechtel)," Rev. 13;
- WO 115689350, "Structural Integrity Test;" and
- 25402-011-V1b-NEE0-00002-001, "Containment Vessel Structural Integrity Test Procedure (Graftel)," dated May 5, 2015.

The inspectors assessed the following attributes associated with the test observation:

- test personnel were on station and had the latest revision of the procedure;
- minimum crew requirements were met;
- test prerequisites were performed;
- testing was executed and coordinated properly;
- testing was performed in accordance with the approved procedures, drawings, and vendor manuals;
- test interruptions and continuations were handled in accordance with approved procedures and documented in the chronological test log;
- testing events and discrepancies were properly documented in the test deficiency log;
- temporary equipment was installed and tracked appropriately; and
- administrative test controls were properly followed.

In addition, the inspectors observed the following SIT and CILRT activities to verify they were in compliance with the applicant's procedure and WO requirements:

- the applicant performed pre-job briefs for multiple shifts and work disciplines. The pre-job briefs discussed critical aspects of the task, safety concerns and precautions, and series of actions that would occur if leaks were identified;
- the reduced pressure test (CILRT) and the peak pressure test (SIT) were performed sequentially during the preoperational CILRT and hold times and pressures were in accordance with required regulations and commitments;
- the containment structure was closed out and a general inspection of the accessible portions of interior and exterior containment surfaces was performed in accordance with procedures;
- the test pressure and hold time used, maximum rate of pressurization, including pressurization increments, and depressurization of the containment were adhered to;
- logging of required test parameters, leakage rates, and observations, which included atmospheric and containment pressure, temperature, dew point temperature, and liquid level (e.g., sump levels, pressurizer level) was performed as required;
- proper system alignment and configuration control for testing criteria was performed in accordance with procedure and work order instructions. The inspectors observed personnel utilizing human error prevention tools to verify they were on the correct component, positioning the valve in the correct line-up, and hanging the tags to ensure configuration control was maintained;
- the test crews were evaluated to verify that the test performance was adequately coordinated and the crews were knowledgeable of the test program and procedures and the application and reading of instrumentation;

- the test results were satisfactory and within acceptance criteria and correctly documented, as required; and
- the inspectors verified that no adverse environmental conditions, such as high wind or extreme temperature differentials, could potentially impact the pressurization and stabilization for the SIT and CILRT.

The inspectors observed the following activities specific to the SIT:

- current calibration records existed for the pressure gauges and temperature sensors used during the SIT. The inspectors verified the type, location, range, accuracy, and calibration of these instruments;
- during the SIT, the applicant verified leak tightness of the steel containment vessel welds added since the original testing of the vessel in 1978. The NRC observed the applicant's examination for leakage at applicable joints and connections. The applicant's examination teams included Bechtel field engineers, quality control inspectors, and Authorized Nuclear Inspectors (ANIs);
- the inspectors observed the pre-test leakage examination performed at 5 pounds per square inch gauge (psig) containment pressure; and
- the inspectors observed the SIT peak pressure of 16.9 to 17.1 psig for one hour, which was greater than the ASME code requirement of ten minutes. The inspectors also observed the depressurization to the final test pressure of 13.5 psig and the subsequent liquid film solution testing of the applicable welds specified by the procedure and/or ASME code.

The inspectors observed the following activities specific to the CILRT:

- the test performed met the requirements of 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," and ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements," which included Type A test requirements; local leak rate testing (LLRT) requirements; instrument calibrations and pretest checks; containment pressurization, depressurization, stabilization, and hold times; verification test; and termination limits;
- the inspectors observed the CILRT pressure of 15.2 to 15.6 psig for a 24-hour hold time. The inspectors also observed the applicant performing the required stabilization hold time and the leak rate verification test. Specifically the inspectors noted that the pressurization source, upon reaching test pressure, was vented and isolated from containment and the pressure-temperature stabilization was observed prior to commencing the leakage-rate measurement;
- the inspectors reviewed the containment area temperature surveys and verified that the instrument strings were operational and adequately covered the volumes covered by the area surveys;
- the inspectors verified that the applicant utilized an NRC accepted method of analysis for the CILRT computer program to determine the overall leak rate. The program was also 10 CFR 50, Appendix B qualified. The inspectors verified that the proper LLRT correction factors were included in the CILRT results and the preliminary CILRT results were satisfactory for the as-left containment conditions;

The final test records review for the SIT/CILRT and the CR closure reviews for 1047373, 1073506, and 1088115 will be performed under the existing inspection program, specifically during IP 70323, "Containment Leak Rate Test Results Evaluation."

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified. Throughout the performance of the SIT/CILRT collection of procedures, several low level issues were identified and entered into the applicant's corrective action program. The resolution of these deficiencies will receive NRC review through the implementation of Section 02.03, Review of Test Records, of IP 63050 and IP 70323, "Containment Leak Rate Test Results Evaluation."

The inspectors noted that the applicant responded to potential leakage paths detected before or during the CILRT. The inspectors observed the applicant utilizing the corrective action program to identify and disposition water identified in the containment raceway and keyway leak chases. These issues were documented under CRs 1047373, 1073506, and 1088115. The inspectors also observed the performance of an LLRT for the raceway leak chase for quadrant 90 to 180 degrees azimuth. The inspectors verified that the results of the LLRT were incorporated into the final CILRT leakage rate and the final results were within the allowable acceptance criteria for the surveillance test.

c. Conclusions

The inspectors determined that the applicant's test procedures and WOs for the SIT and CILRT were performed in a manner consistent with the guidance of procedure SMP-9, Watts Bar Nuclear Plant Unit 2, "Conduct of Test," Rev. 6. With respect to IP 70313, this IP was performed in its entirety. However, only section 02.02 of IP 63050 was performed during this inspection. Inspection of IP 63050, Section 02.01, was documented in Section P.1.3 of this report. Further inspection of Section 02.03, Review of Test Records is required for IP 63050.

P.1.9 Preoperational Test Witnessing (Inspection Procedures 70315 and 70434B)

a. Inspection Scope

Background: The background for this preoperational test witnessing is the same as that in the background section of P.1.5.

Inspection Activities: The inspectors witnessed activities associated with the performance of Preoperational Test Instruction 2-PTI-099-03, "Reactor Protection System Operational Check," Rev. 1 to verify that the testing was conducted in accordance with approved procedures and to verify the adequacy of test program records and preliminary evaluation of test results. The following test sections were selected for inspection of this item:

- Section 6.1, Manual Reactor Trip and Bypass Breaker Operations and Trip
- Section 6.3, Permissives P-7, P-10 and P-13
- Section 6.4, Permissive P-8

- Section 6.5, Permissive P-9
- Section 6.9, Power Range High Flux (High Setpoint) Reactor Trip
- Section 6.11, Overtemperature Delta T Reactor Trip
- Section 6.12, Overpower Delta T Reactor Trip
- Section 6.13, Reactor Coolant Low Flow Reactor Trip
- Section 6.14, Pressurizer High Pressure Reactor Trip
- Section 6.15, Pressurizer Low Pressure Reactor Trip
- Section 6.16, Pressurizer High Level Reactor Trip

The inspectors assessed the following attributes associated with this test observation:

- all test personnel were on station and had the latest revision of the procedure;
- test prerequisites were performed;
- plant systems were in service to support the test;
- minimum crew requirements were met;
- testing was performed in accordance with the approved procedure;
- test interruptions and continuations were handled in accordance with approved procedures and documented in the chronological test log;
- testing events and discrepancies were properly documented evaluated and in the test deficiency log;
- testing was executed and coordinated properly;
- data was properly collected;
- temporary equipment was installed and tracked appropriately;
- administrative test controls were properly followed; and
- test personnel were using approved drawings and vendor manuals.

The inspectors observed the tests to verify that the overall test acceptance criteria were met. The inspectors conducted a review with the responsible test engineer to assure that the preliminary test evaluations were consistent with the inspector's observations. During the tests, the inspectors observed important data gathering activities to ensure the data was properly gathered and recorded. A post-test cursory review of the test data was performed to verify legibility, traceability, and permanence of the data sheet entries.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors determined that the applicant's test procedure was performed in a manner consistent with the guidance of procedure SMP-9, Watts Bar Nuclear Plant Unit 2, "Conduct of Test," Rev. 6.

P.1.10 Preoperational Test Witnessing (Inspection Procedures 70315 and 70434B)

a. Inspection Scope

Background: The background for this preoperational test witnessing is the same as that in the background section of P.1.5.

Inspection Activities: The inspectors witnessed activities associated with the performance of Preoperational Test Instruction 2-PTI-099-04, "Safeguards System," Rev. 1 to verify that the testing was conducted in accordance with approved procedures and to verify the adequacy of test program records and preliminary evaluation of test results. The following test sections were selected for inspection of this item:

- Section 6.1, Manual Safety Injection Train A
- Section 6.2, Manual Containment Spray Actuation Train A
- Section 6.3, Containment Hi-Hi Pressure Steam Line Isolation
- Section 6.4, Pressurizer Pressure P-11
- Section 6.5, Low Pressurizer Pressure Safety Injection
- Section 6.6, High Steam Pressure Rate Steam Line Isolation
- Section 6.7, Containment Hi Pressure Safety Injection
- Section 6.8, Low Steamline Pressure Safety Injection
- Section 6.11, Low Tavg/P-4 Feedwater Isolation

The inspectors assessed the following attributes associated with this test observation:

- all test personnel were on station and had the latest revision of the procedure;
- test prerequisites were performed;
- plant systems were in service to support the test;
- minimum crew requirements were met;
- testing was performed in accordance with the approved procedure;
- test interruptions and continuations were handled in accordance with approved procedures and documented in the chronological test log;
- testing events and discrepancies were properly evaluated and documented in the test deficiency log;
- testing was executed and coordinated properly;
- data was properly collected;
- temporary equipment was installed and tracked appropriately;
- administrative test controls were properly followed; and
- test personnel were using approved drawings and vendor manuals.

The inspectors observed the tests to verify that the overall test acceptance criteria were met. The inspectors conducted a review with the responsible test engineer to assure that the preliminary test evaluations were consistent with the inspector's observations. During the tests, the inspectors observed important data gathering activities to ensure the data was properly gathered and recorded. A post-test cursory review of the test data was performed to verify legibility, traceability, and permanence of the data sheet entries.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors determined that the applicant's test procedure was performed in a manner consistent with the guidance of procedure SMP-9, Watts Bar Nuclear Plant Unit 2, "Conduct of Test," Rev. 6.

P.1.11 Preoperational Test Witnessing (Inspection Procedures 70312 and 70315)

a. Inspection Scope

Background: The background for this preoperational test witnessing is the same as that in the background section of P.1.5.

Inspection Activities: During this inspection, inspectors witnessed preoperational tests performed based on approved test procedures 2-PTI-262-01, Rev. 0001, "Unit 2 Integrated Safeguards Test – Train 2A," and 2-PTI-262-02, Rev. 0001, "Unit 2 Integrated Safeguards Test – Train 2B." The inspectors witnessed a portion of each section of both tests. Inspectors interviewed responsible individuals conducting and directing the test during and following the performance of the different test sections to confirm that testing was being conducted in accordance with established administrative controls in the startup manual procedures and to understand and review test results. The inspectors verified that all identified test deficiencies were properly documented in TDNs and/or CRs as appropriate for problem evaluation, correction, or retesting.

b. Findings and Observations

Introduction: A self-revealing, SL IV, NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified when the applicant failed to provide adequate directions for work scope nor specify a rigorous post modification test to ensure that all affected equipment was returned to an operable status.

Description: On September 11, 2015, while performing 2-PTI-262-02, Section 6.1.2, Step 70, "Performance of Loss of Offsite Power (LOOP)," it was observed that certain loads did not strip from the bus as required by acceptance criteria. According to the Updated Final Safety Analysis Report (UFSAR) Table 8.3-2, "Shutdown Board Loads Automatically Tripped Following a Loss of Preferred (Offsite) Power," a number of nonsafety-related loads connected to safety-related power distribution boards were required to strip and not automatically reconnect to their normal power supply. During the test observation, the following loads did not strip:

- Primary Water Makeup Pump 2B (2-PMP-81-7), 0-MTR-40-66B, 2-BKR-81-7, C/2D;
- Valve & Strainer Room Sump Pump B (0-MTR-40-66B), 0-BKR-40-66B, C/2F1;
- Boric Acid Batch Tank Heater 3 (2-HTR-62-228/3), 2-BKR-62-228/3, C/17A;
- Diesel Generator Engine 2B1 Immersion Heater (2-HTR-82-B1), 2-BKR-82-B1/2, C/2C;
- Diesel Generator Engine 2B2 Air Compressor, 2-BKR-82-271, Compartment 5A;
- Diesel Generator Building Corridor Heater (0-HTR-30-482), 0-BKR-30-482, Compartment 5F1;
- Diesel Aux Board 2B1-B Power Outlets (2-PO-215-5,6,7), 2-BKR-215-B001/5 F2, Compartment 5F2;
- Instrument Room A/C Compressor 2B (2-COMP-31-324/2), 2-MTR-31-324B; 2-BKR-31-324B, C/17F2;

- Reactor MOV Board 2B1-B Power Outlets (2-PO-213-B1/1, 2, 3, 4, 5), 2-BKR-213-B001, 16E;
- Reactor MOV Board 2B1-B Power Outlets (2-PO-213-B1/6, 7, 8, 9, 10), 2-BKR-213-B001, Compt. 17E;
- Boric Acid Batch Tank Heater 3, 2-BKR-62-228, 3; and
- Lower Compartment Cooler 2B-B (2-CCU-30-75), 2-BKR-30-75, Compartment 7D.

One additional equipment item was impacted by the load strip failure, but was not included in the scope of PTI-262-02:

- Containment Annulus Vacuum Fan; 2-MTR-65-77.

The test anomaly was documented in TDN 15-1638 and CR 1081117. An interview with the startup test supervisor indicated that the problem with the failure to strip loads was a discontinuity in wiring to the coil of the logic relay. On April 3, 2015, during the performance of WO 116522173, "SUTE System 212 DCN 59164," workers disconnected a wire in order to abandon in place four load shed timer delay relays. A part of this scope of work was to lift and re-land wire DCE2 as indicated in item 3 of the MAI-3.3 data sheet record. However, the craft personnel failed to land the wire back on its terminal point, leaving the load shed circuits controlled by relay 2-27X-212 inoperable. The functionality of the circuitry was not adequately tested following completion of the modification because the approved work instructions did not prescribe an adequate post modification test.

The applicant's failure to include adequate post modification testing to verify circuit integrity in the approved work instructions was a performance deficiency. The inspectors determined that the performance deficiency was more than minor because it represented an inadequate process or procedure that if left uncorrected, could adversely affect testing of safety-related SSCs, in accordance with Appendix C, Minor Violations and Findings, of IMC 2517, "Watts Bar Unit 2 Construction Inspection Program," dated June 6, 2013. The inspectors reviewed this finding against cross-cutting area components as described in IMC 0310, "Components Within the Cross-Cutting Areas," and determined that no cross-cutting aspect applied.

Enforcement: 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires 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. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, on April 3, 2015, during the performance of WO 116522173, the applicant failed to provide adequate work directions for performance of work of a type appropriate to the circumstances, and failed to specify appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the applicant failed to include adequate post modification testing to verify circuit integrity in the approved work instructions.

This finding was determined to be a SL IV violation consistent with Section 6.5 of the Enforcement Policy because the applicant failed to establish adequate controls over

construction or testing processes that are important to safety. Because this was a SL IV violation, and the issue was entered into the applicant's corrective action program as CR 1081117, this violation is being treated as an NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000391/2015608-03; "Failure to provide adequate instructions for plant modifications"

c. Conclusion

The inspectors concluded that overall, preoperational tests 2-PTI-262-01 and 02 were conducted in accordance with established administrative controls in the startup manual procedures. Test deficiencies were documented in TDNs and CRs where appropriate. The applicant plans to determine the cause of any problems and correct any equipment performance deficiencies and conduct retesting in a subsequent PTI 262-03 which will be subject to future NRC inspections.

P.1.12 Preoperational Test Results Evaluation (Inspection Procedure 70400)

a. Inspection Scope

Background: IMC 2513, "Light Water Reactor Inspection Program - Preoperational Testing and Operational Preparedness Phase," issue date January 1, 1984, purpose is to verify through direct observation, personnel interviews, and review of facility records that:

- systems and components important to the safety of the plant are fully tested to demonstrate that they satisfy their design requirements.
- management controls and procedures, including QA programs, necessary for operation of the facility have been documented and implemented.

IMC 2513 defines the inspection program that supports the issuance of an Operating License. IMC 2513 requires the preoperational test results review of the mandatory tests defined in IMC 2513 and five of the primal tests defined in IMC 2513. The following inspection was performed to in relation to satisfying the required preoperational test results review.

Inspection Activities: The inspectors performed a detailed review of the results for preoperational test procedures 2-PTI-067-04, "Essential Raw Cooling Water (ERCW) Dual Unit Shutdown Flow Test," Rev. 0, 2-PTI-067-02A, ERCW System Flow Balance – Train A," Rev.4, and 2-PTI-067-02B, "ERCW System Flow Balance – Train B, Rev. 2, and WO 116923934, "Collect CCS Train B Flow Data With Both CCS Pumps In Service," to verify that the applicant's evaluation of the procedure performance and results was conducted in accordance with approved procedures. This review was performed to provide assurance that the test data was within the established acceptance criteria and the applicant's methods for identifying and correcting deficiencies were adequate. The inspectors performed the following activities associated with this test results review:

- reviewed all changes made to the test procedure to verify they were properly annotated, did not affect the objective of the test, and were performed in accordance with administrative procedures;

- reviewed all documented test deficiencies to verify they had been properly resolved, reviewed, and accepted;
- reviewed the test summary and evaluation to verify that the system was evaluated to meet design requirements and acceptance criteria;
- reviewed the original “as-run” copy of the test to verify completion of data sheets, calculations, and signatures/initials; and
- the approval of the test results were reviewed for completeness to ensure that personnel charged with the responsibility for review and acceptance had documented their evaluation and corrected any identified discrepancies.

The inspectors reviewed the test results to verify that the overall test acceptance was met. The inspectors conducted a review with the responsible test engineer to assure that the test evaluation was performed in accordance with established procedures.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors determined that the applicant’s test results were processed in a manner consistent with the guidance of procedure SMP-10.0, “Watts Bar Nuclear Plant Unit 2 Packaging and Processing Test Results,” Rev. 2. This completes the test results evaluation of preoperational test procedures 2-PTI-067-02A, 2-PTI-067-02B, 2-PTI-067-04, and WO 116923934, “Collect CCS Train B Flow Data with both CCS Pumps.

P.1.13 Preoperational Test Results Evaluation (Inspection Procedure 70400)

a. Inspection Scope

Background: The background for this preoperational test results evaluation is the same as that in the background section of P.1.12.

Inspection Activities: The inspectors performed a detailed review of the results for the following preoperational test procedures:

- 2-PTI-001-01, “Main Steam Isolation Valves and Bypass Isolation Valves,” Rev. 1;
- 2-PTI-003B-04, “Auxiliary Feedwater Pumps & Valves Logic Test,” Rev. 1;
- 2-PTI-003B-05, “Auxiliary Feedwater System Dynamic Test,” Rev. 1;
- 2-PTI-062-02, “Chemical Volume and Control Boric Acid Subsystem Logic Testing,” Rev. 1;
- 2-PTI-063-03, “Safety Injection System Charging, SI, and RHR Flow Balance Test,” Rev. 2;
- 2-PTI-063-05A, “Safety Injection System Pump/Valve Logic Test - Train A,” Rev. 2;
- 2-PTI-063-05B, “Safety Injection System Pump/Valve Logic Test - Train B,” Rev. 2;

- 2-PTI-063-06, "Safety Injection System Check Valve Test," Rev.1;
- 2-PTI-070-01, "Component Cooling Water Pump/Valve Logic Test," Rev.1;
- 2-PTI-070-02A, "Component Cooling System Unit 2 Train A Flow Balance," Rev.1;
- 2-PTI-070-02B, "Component Cooling System Unit 2 Train B Flow Balance," Rev.1;
- 2-PTI-074-02A, "RHR Midloop Operation," Rev. 1; and
- 2-PTI-074-02B, "RHR HFT Heatup/Cooldown," Rev. 1.

These reviews were implemented to verify that the applicant's evaluation of the procedure performance and results were conducted in accordance with approved procedures. This review was performed to provide assurance that the test data was within the established acceptance criteria and the applicant's methods for identifying and correcting deficiencies were adequate. The inspectors performed the following activities associated with these test results reviews:

- reviewed all changes made to the test procedure to verify they were properly annotated, did not affect the objective of the test, and were performed in accordance with administrative procedures;
- reviewed all documented test deficiencies to verify they had been properly resolved, reviewed, and accepted;
- reviewed the test summary and evaluation to verify that the system was evaluated to meet design requirements and acceptance criteria;
- reviewed the original "as-run" copy of the test to verify completion of data sheets, calculations, and signatures/initials;
- QA inspection records were reviewed to verify they were completed as required by the test procedure; and
- the approval of the test results were reviewed for completeness to ensure that personnel charged with the responsibility for review and acceptance had documented their evaluation and corrected any identified discrepancies.

The inspectors reviewed the test results to verify that the overall test acceptance was met. The inspectors conducted a review with the responsible test engineer to assure that the test evaluation was performed in accordance with established procedures.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors determined that the applicant's test results were processed in a manner consistent with the guidance of procedure SMP-10.0, "Watts Bar Nuclear Plant Unit 2 Packaging and Processing Test Results," Rev. 2. This completes the test results evaluation of the aforementioned preoperational test procedures.

P.1.14 Preoperational Test Results Evaluation (Inspection Procedure 70400)

a. Inspection Scope

Background: The background for this preoperational test results evaluation is the same as that in the background section of P.1.12.

Inspection Activities: The inspectors performed a detailed review of the results for preoperational test procedure 2-PTI-099-05, "Overpower Delta-T and Overtemperature Delta-T Turbine Runback", Rev. 1 to verify that the applicant's evaluation of the procedure performance and test results was conducted in accordance with approved procedures. This review was performed to provide assurance that the test data was within the established acceptance criteria and the applicant's methods for identifying and correcting deficiencies were adequate. The inspectors performed the following activities associated with this test results review:

- Reviewed all changes made to the test procedure to verify they were properly annotated, did not affect the objective of the test, and were performed in accordance with administrative procedures;
- Reviewed all documented test deficiencies to verify they had been properly resolved, reviewed, and accepted;
- Reviewed the test summary and evaluation to verify that the system was evaluated to meet design requirements and acceptance criteria;
- Reviewed the original "as-run" copy of the test to verify completion of data sheets, calculations, and signatures/initials; and
- The QA inspection records were reviewed to verify they were completed as required by the test procedure.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors have not yet determined if the applicant's test data meets the established acceptance criteria. Additional inspection will need to be performed in order to evaluate the test results of preoperational test procedure 2-PTI-099-05 before closure of the item.

P.1.15 (Discussed) Control of Radioactive Materials and Contamination, Surveys, and Monitoring (Preoperational and Supplemental) (Inspection Procedure 83526)

a. Inspection Scope

The inspectors walked-down U2 Area Radiation Monitors (ARMs) and airborne radioactivity monitors (System 90) and assessed the components for material condition and conformance with system design documents including the FSAR. The walk downs included observations of the following System 90 components that are unique to U2:

- Personnel Air Lock ARM, 2-RE-90-2
- Containment Refueling Floor, 2-RE-90-59
- Containment Refueling Floor, 2-RE-90-60

- Lower Compartment Instrument Room, 2-RE-90-61
- Reactor Building Upper Compartment Post Accident ARM, 2-RE-90-271/272
- Reactor Building Lower Compartment Post Accident ARM, 2-RE-90-273/274
- Reactor Building Particulate, Iodine, Noble Gas, 2-RE-90-106/112

The inspectors discussed radiation monitor function with applicant staff.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified concerning radioactive material control and radiation monitoring activities. The inspectors noted that Unit 1 capabilities currently in place for portable surveys, protective clothing, contamination control, and exit-point monitoring are adequate to support dual-unit operation. The inspectors observed that all the ARMs have been installed with the exception of 2-RE-90-2, which is only partially complete. The inspectors also observed that 2-RE-90-106/112 has been installed, except for the heat tracing on the sample lines outside containment. The inspectors noted that none of the radiation monitors within the scope of this IP have had preoperational testing completed.

c. Conclusions

This IP could not be completed because none of the radiation monitors (System 90) have had preoperational testing completed. Future inspections will focus on the review of preoperational testing records for the above listed components and verification of complete installation for 2-RE-90-2 and 2-RE-90-106/112.

**P.1.16 (Discussed) Liquids and Liquid Wastes (Preoperational and Supplemental)
(Inspection Procedure 84523)**

a. Inspection Scope

The inspectors walked-down accessible sections of the U2 liquid radioactive waste processing system (System 77) and U2 liquid effluent monitoring system (System 90) and assessed the components for material condition and conformance with system design documents including the FSAR. The walkdowns included observations of the following System 77 and System 90 components that are unique to U2:

- Steam Generator Blowdown Effluent Monitor, 2-RE-90-120/121
- Reactor Coolant Drain Tank (RCDT) to Gas Analyzer Flow Control Valve, 2-FCV-77-16
- RCDT to Vent HDR Flow Control, 2-FCV-77-18
- RCDT Pump Discharge VLV Flow Control, 2-FCV-77-9
- RCDT Pump Discharge Flow Control, 2-FCV-77-10
- RCDT to Gas Analyzer Flow Control, 2-FCV-77-17
- RCDT to Vent Header Flow Control, 2-FCV-77-19
- RCDT N2 Supply Flow Control, 2-FCV-77-20

The inspectors discussed component function and liquid radwaste program implementation with applicant staff. The inspectors also reviewed preoperational testing and transfer calibration records associated with 2-RE-90-120/121.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified concerning U2 liquid radwaste system preoperational activities. The inspectors previously determined that the bulk of the U2 liquid radwaste processing system is shared with U1 and has been successfully operating since U1 startup. The inspectors also noted that System 77 components were designed and built for dual-unit operational capacities. For the System 77 components unique to U2, the inspectors observed, on a smart-sampling basis, that the components had been installed and construction appeared to be complete. The inspectors noted that preoperational testing of this system has not been completed.

The inspectors performed a walk-down of 2-RE-90-120/121 and noted that installation of the monitoring skid has been completed, along with associated sample lines, valves, pumps, flowmeters, and heat exchangers. The inspectors also noted that preoperational testing for this radiation monitor has been completed, including loop calibrations and automatic valve actuation testing.

c. Conclusions

This IP could not be completed because several System 77 components have not yet had preoperational testing completed. Future inspections in this area should include an evaluation of preoperational testing records for the RCDT, containment sumps, and associated pumps and valves (including those relied upon to isolate containment); to the extent that U2 System 77 can reasonably be verified as operable. For purposes of this IP, no additional inspection activities are required for System 90 liquid effluent monitoring components.

**P.1.17 (Discussed) Gaseous Waste System (Pre-Operational and Supplemental)
(Inspection Procedure 84524)**

a. Inspection Scope

The inspectors walked-down selected accessible portions of U2 Gaseous Waste Disposal System (GWDS) (System 77), the U2 gaseous effluent monitoring system (System 90), Reactor Building Purge Ventilating System (RBPVS) (System 30), and the Containment Purge Ventilation System (System 30). Equipment and components were assessed for material condition and conformance with system design documents including the FSAR. The walkdowns included observations of the following System 30, System 77, and System 90 components that are unique to U2 (along with associated pumps, valves, and piping):

- Containment Purge Air Exhaust Effluent Monitors (2-RE-90-130/131)
- Containment Purge Air Exhaust Filter Units
- Shield Building Exhaust Vent

The inspectors discussed component and system function and preoperational testing program implementation with applicant staff. The inspectors also reviewed design basis setpoint determination calculations for selected radiation monitors.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified concerning U2 gaseous radwaste system preoperational activities. The inspectors previously determined that the bulk of the U2 gaseous radwaste processing system is shared with U1 and has been successfully operating since U1 startup and that System 77 components were designed and built for dual-unit operational capacities. For the System 77 components unique to U2, the inspectors observed that the components had been installed and construction appeared to be complete. The inspectors noted that preoperational testing of this system has not been completed.

All of System 90 gaseous effluent monitors have not been turned-over to Startup Engineering, however the inspectors observed that installation/construction of 2-RE-90-130/131 (containment purge exhaust) appeared to be complete. Installation of three additional flow elements for Shield Building Exhaust Vent flow monitoring (U2 Containment Purge Exhaust Train A & B and Emergency Gas Treatment System (EGTS) Train B) for design change notice (DCN) 54207-A has been completed, however, preoperational testing has identified problems with signal cabling.

The Auxiliary Building Gas Treatment System ventilation is shared with U1 and required for U1 operation. The RBPVS and EGTS are U2 specific. Preoperational testing and the system flow balance are to be performed using simulated filtration. The inspectors did not review preoperational testing of these systems because charcoal and High-efficiency Particulate Air (HEPA) filters will be installed and tested just prior to fuel load.

c. Conclusions

This inspection procedure could not be completed because several components have not yet had preoperational testing performed. Future inspections in this area should include a review of preoperational testing records related to Systems 30, 65, and 77. The review should include operability of GWDS pumps and valves (including those relied upon to isolate containment), in-place testing of charcoal/HEPA air cleaning systems after filtration media is installed, and operability of additional flow elements for Shield Building Exhaust Vent flow monitoring. Additional inspection activities for System 90 components should include a review of 2-RE-90-130/131 preoperational testing results.

III. OPERATIONAL READINESS ACTIVITIES

O.1 Operations

O.1.1 (Closed) Operating Procedures (Inspection Procedure 42450B)

a. Inspection Scope

Background: Previously, a review of General Operating Instructions, Operating Test Instructions, System Operating Instructions (SOIs), Technical Instructions (TIs), and Technical Requirements Instructions (TRIs) was documented in IIR 05000391/2014605 (ADAMS Accession No. ML14226A049), Section O.1.2.

Inspection Activities: The inspectors reviewed a sample of the surveillance instructions and compared them with the existing Unit 1 procedures, using criteria in:

- ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants;"
- ANSI N18.7-1976/ANS 3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants;"
- NRC Regulatory Guide (RG) 1.33-1978, Rev. 2, "Quality Assurance Program Requirements (Operation);"
- RG 1.33-2013, Rev. 3, "Quality Assurance Program Requirements (Operation);"
- and
- TVA-NQA-PLN89-A, "Quality Assurance Program Description."

Each inspection activity was performed to verify that plant operating procedures were prepared to adequately control safety related operations within the applicable regulatory requirements, to verify that differences between the units were incorporated into Unit 2 procedures, and to verify that the procedures met the criteria in RG 1.33. Inspectors obtained copies of the indices for the plant's surveillance instructions and reviewed them for completeness.

The inspectors reviewed and compared a sampling of Unit 1 and Unit 2 surveillance instructions to verify whether or not unit differences were incorporated, and to verify whether or not the procedures met the criteria specified in RG 1.33.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors determined that the Watts Bar active and in-process Unit 2 surveillance instructions were prepared to adequately control safety-related operations in accordance with specific regulatory requirements. IP 42450B is closed.

O.1.2 (Closed) Technical Specifications Review (Inspection Procedure 71301)

a. Inspection Scope

Background: The results of previous inspection activities were documented in Watts Bar Nuclear Plant Unit 2 Construction IIR 05000391/2015605 (ADAMS Accession No. ML15226A345) and 05000391/2015607 (ADAMS Accession No. ML15273A452). The scope of this inspection was limited to follow-up inspection activities for the items listed in the 05000391/2015607 IIR.

Inspection Activities: The inspectors verified that:

- 2-SI-0-4, Monthly Surveillances, Appendix D, Remote Shutdown and PAM Channel Check Data, Items 11 and 12, matched the post-accident monitor screen displays on control room panels 2-M-4 and 2-M-6, to ensure the intent of Tech Spec surveillance requirement SR 3.3.3.1 was met with 2-SI-0-4, Appendix D.
- 0-SI-65-6-A/B, Emergency Gas Treatment System Train A/B 10-Hour Operation, included guidance to alternate the EGTS units' suction/discharge paths between the Unit 1 and Unit 2 containment annulus areas with some periodicity, to ensure that the intent of Tech Spec surveillance requirement SR 3.6.9.1 was met with 0-SI-65-6-A/B.
- 0-SI-82-5, 18 Month Loss Of Offsite Power With Safety Injection Test - DG 2A-A, and 0-SI-82-6, 18 Month Loss Of Offsite Power With Safety Injection Test - DG 2B-B, included verifications of all ERCW and CCS pump/valve automatic alignments during a safety injection signal and during a loss of offsite power, to ensure that the intent of Tech Spec surveillance requirements SR 3.7.7.4, SR 3.7.8.2, and SR 3.7.8.3 were met with 0-SI-82-5 and 0-SI-82-6.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

Based on the aforementioned inspection activities, no further inspection is required. IP 71301 is closed.

IV. OTHER ACTIVITIES

OA 1.1 (Discussed) Generic Letter 89-04: Guidance on Developing Acceptable In-Service Testing Programs and Temporary Instruction 2515/114: Inspection Requirements for Generic Letter 89-04, Acceptable In-Service Testing Programs

a. Inspection Scope

Background: Generic Letter (GL) 89-04 informed licensees of NUREG-1482, "Guidelines for Inservice Testing Programs at Nuclear Power Plants." NUREG-1482 contained recommendations for developing and implementing in-service testing (IST) programs. NUREG-1482 referenced paragraph 50.55.a(b) of 10 CFR, which stated the requirements of the ASME Operation and Maintenance for Nuclear Power Plants Code was incorporated by reference into the NRC regulations. Paragraph 50.55a(f)4 of 10 CFR requires that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves which are classified as ASME Code Class 1,2, and 3 must meet the in-service test requirements of ASME OM Code as incorporated by reference in 10 CFR 50.55a(b). As referenced in SSER 22 Section 3.9.6, TVA was expected to submit an IST program and NRR open specific relief requests for WBN Unit 2 nine months before the projected date of operating license

issuance. TVA submitted a letter, dated December 12, 2013, which provided the NRC staff with TVA's Technical Instruction (TI), 0-TI-100-006, "Inservice Testing Program" for the WBN Units 1 and 2, Rev. 0. For Unit 1 WBN submitted Enclosure 1, which was an update to the IST program to the staff for the Unit 1 third 10-year interval. Enclosure 2 of the submittal requested to allow (1) alignment of the 120 month interval dates for the Unit 1 IST program to be concurrent with Unit 2 IST program, and (2) Unit 1 and Unit 2 to utilize the latest edition and addenda of the ASME OM Code currently reference by 10 CFR 50.55a(b), which is ASME OM Code 2004 Edition through 2006 Addenda.

Inspection Activities: The inspectors reviewed the preservice test procedure for the 2A-A and 2B-B safety injection (SI) pump tests. In addition, the inspectors observed the preservice test for SI pump 2A-A, and reviewed the test records for both pumps to verify that the tests were completed in accordance with the approved test procedure, acceptance criteria was established and met. In addition, the inspection was completed to verify the test instrumentation was adequate and requirements of ASME OM Code 2004 Edition through 2006 Addenda were met.

TI 2515/114 samples completed:

- 03.02 b, d, g, and h – two samples
- 03.03 a and b – two samples
- 03.06, a, b, c, d, and e – two samples

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The preservice test for the 2A-A and 2B-B safety injection pumps was completed in accordance with the approved procedures and met the requirements of ASME OM Code 2004 Edition through 2006 Addenda.

**OA.1.2 (Discussed) Instrument Components and Systems – Work Observation
(Inspection Procedure 52053)**

a. Inspection Scope

Background: As discussed in Section C.1.2, the reconstitution process was documented in IIR 05000391/2009602, Attachment 2, and included the results for IP 52053. The reconstitution results stated that the inspection requirements were not met and IP 52053 should be performed in its entirety. IP 52053 was closed in IIR 05000391/2015605 (ADAMS Accession No. ML15226A345).

Inspection Activities: The inspectors observed calibration work performed under surveillance instruction 2-SI-92-42, "18 Month Channel Calibration of Power Range Nuclear Instrumentation System Channel N-42," Rev. 1. For the testing and calibration, the inspectors verified that the (1) latest revisions of applicable procedures were available at the work location and used by qualified personnel performing the testing; (2)

calibration and M&TE used was properly identified, traceable, and calibrated; (3) components calibrated were able to obtain the set point and within the tolerance specified; (4) testing and calibration results were recorded during the activity; and (5) components were adequately identified.

The following samples were completed:

- Section 02.02.f – one sample

b. Observations and Findings:

No findings were identified.

c. Conclusion:

The nuclear instrumentation calibration was complete in accordance with the approved procedures. Additional activities may be performed at the NRC's discretion.

OA.1.3 (Discussed) Radiation Monitoring System Special Program (Temporary Instruction 2512/041)

a. Inspection Scope

Background: In 1989, the Watts Bar NPP established a Special Program (SP) to provide programmatic corrective actions for deficiencies with the design, documentation, installation, and application of hardware used in the liquid and gaseous radiation monitoring systems (RMS). The SP provided actions to address three primary concerns; specifically, (1) ensure that criteria in design-basis documents include applicable requirements of Regulatory Guide 1.97, "Post Accident Monitoring; (2) evaluate the RMS design, documentation, and installation against the updated design criteria to verify acceptable installation; and (3) ensure installation deficiencies are modified or reworked and are documented to show correction of the deficiencies. A 2010 inspection, IIR 05000391/2010603 (ADAMS Accession No. ML102170465), confirmed that the required design criteria had been established. The most recent inspection of this SP was documented in IIR 05000391/2011607 (ADAMS Accession No. ML112730197). That inspection examined vendor calibrations of radiation detectors and determined that the associated documentation met applicable requirements and provided sufficient information to establish traceability to primary standards.

Inspection Activities:

Radiation Sampling Systems

In this inspection, inspectors evaluated as-installed configurations for the following skid-mounted equipment and associated sample piping:

- 2-RE-90-106, Containment Lower Compartment Particulate & Noble Gas Monitor,
- 2-RE-90-112, Containment Upper Compartment Particulate & Noble Gas Monitor,
- 2-RE-90-119, Condenser Vacuum Exhaust Low Range monitor,

- 2-RE-90-120 and 121, Steam Generator Blowdown monitors, and
- 2-RE-90-130 and 131, Containment Purge Air Exhaust Monitors.

To evaluate the installations, the inspectors conducted interviews with responsible applicant personnel, examined configuration drawings and specifications, reviewed equipment setpoint calculations, and walked down physical installations of the equipment. The inspection included an evaluation of whether there has been a recurrence of historical issues, including insufficient bend radii in piping, deficiencies in sample flow control, blockages and restrictions in sample lines, excessive length of piping, incorrect sloping of piping, inadequate heat tracing, use of incompatible piping material (e.g. copper), wiring errors, pipe caps not installed, and spurious actuations from signal transients.

Area Radiation Monitors

The inspectors evaluated as-installed configurations for the following area monitors:

- 2-RE-90-02, Containment Personnel Airlock Access Point;
- 2-RE-90-59 and 60, Refueling Deck; and
- 2-RE-90-61, Seal Table.

The inspectors examined configuration drawings and specifications, reviewed equipment setpoint calculations, and walked down physical installations. The review was performed to determine if there has been a recurrence of historical deficiencies, such as lack of analysis for placements, issues with mounting details, and damage to cables and electrical connections.

Setpoints for Radiation Monitors

The inspectors evaluated whether there has been a recurrence of a historical concern where inconsistencies had existed between established and documented setpoints for control room annunciators. This inspection was accomplished by verifying setpoint values defined by engineering had been flowed down into calibration procedures for the following radiation monitors:

- 2-RE-90-106 and 112, Containment Upper and Lower Compartment Particulate & Noble Gas Monitors;
- 2-RE-90-120 and 121, Steam Generator Blowdown monitors;
- 2-RE-90-130 and 131, Containment Purge Air Exhaust Monitors;
- 2-RE-90-119, Condenser Vacuum Pump Exhaust Continuous Monitor (Low Range);
- 2-RE-90-255, and 256, Condenser Vacuum Pump Exhaust Monitor (Mid and High Range);
- 2-RE-90-271, 272, 273, and 274, Containment High Range Radiation Monitors; and
- 2-RE-90-421, 422, 423, and 424, Main Steam Line Monitors.

The inspectors reviewed setpoint and scaling documents (SSDs) and engineering calculations to verify specified setpoints were accurately incorporated into procedures, setpoints conformed to the Offsite Dose Calculation Manual (ODCM), values for gas monitors addressed effects of transmission factor and particle loss in sample lines, and

monitoring systems were capable of detecting a 1 gallon per minute (gpm) reactor coolant system leak.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

Inspection of radiation sampling systems yielded the following observations:

- In general, the inspected piping installations that had been completed did not exhibit issues with design or workmanship. Some piping assembly and installations of pipe caps were still in progress for the RE-106 and 112 equipment skids.
- Installation of required heat tracing was still pending for the sample piping for monitors RE-106 and 112.
- Monitors RE-106, 112, 119, 130, and 131 had not yet been energized or tested to demonstrate adequacy of wiring for equipment control and functionality. Functional testing had been completed only for RE-120 and 121.
- The inspectors' review of engineering document construction releases (EDCRs) and system specifications for radiation monitoring systems determined that the equipment design provided measures to protect against inductive switching surges and electromagnetic or radio frequency interferences.
- With the exception of the Steam Generator Blowdown radiation monitoring system (RE-120 and 121), sample lines for radiation monitors had not yet been tested to demonstrate freedom from flow restrictions or blockages.

Inspections of the area radiation monitors yielded the following observations:

- Installation of 2-RE-90-02 was still in progress and was awaiting installation of the detector.
- None of the area monitors examined in this inspection scope had yet been energized or tested to demonstrate adequacy of wiring for equipment control and functionality.

The overall review of RMS instrument and alarm settings yielded the following observations:

- The inspectors determined that calibration and alarm settings for the containment purge exhaust monitors implemented values specified in engineering SSDs and were consistent with the ODCM.
- The inspectors determined that Technical Instruction TI-18, Calculation Methods for Effluent Radiation Monitors, was established to implement the ODCM requirements for effluent releases and maintain compliance with the Tech Spec offsite dose limits. However, a revision to the technical instruction was still pending to provide information for Unit 2 monitors.
- The inspectors determined that controls for establishment of alarm setpoints for effluent radiation monitors required the use of "Open EMS" software. The software had not yet been updated to include Unit 2 radiation monitor specific

information. Information was still needed from Engineering in order for Chemistry to make the software changes.

- Technical requirements were still pending to define setpoint requirements for the main steam line radiation monitors.

c. Conclusion

The inspectors determined that further inspection is required to verify measures implemented under the SP for radiation monitoring systems have progressed sufficiently to demonstrate the historical concerns have been addressed.

OA.1.4 (Discussed) Three Mile Island Action Item II.F.1: Noble Gas Effluent Monitors (Inspection Procedure 35007)

a. Inspection Scope

Background: As part of the NRC response to the accident at Three Mile Island (TMI), the agency issued NUREG-0737, "Clarification of TMI Action Plan Requirements." Section II.F.1 established a requirement to install noble gas monitors that have the capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident.

The previous inspection of this TMI action item for Unit 2 was documented in IIR 05000391/2013607 (ADAMS Accession No. ML13273A512). That report identified that the installation of the externally mounted high and mid-range detectors (2-RE-255 and 256) and the skid mounted sampling system (2-RE-90-119) for the condenser vacuum exhaust effluent path was still in progress and had not been completed. The main steam line monitors also had not been installed at the time of that inspection.

Inspection Activities: In this inspection, the inspectors conducted direct observations of completed installations, interviewed responsible personnel, and reviewed completed work records to evaluate the actions taken to address the TMI action items for noble gas monitors. The inspectors examined hardware designs and conducted walk downs of hardware installations to evaluate the as-built configurations. Inspectors reviewed corrective actions associated with the mounting and installation of the radiation detectors to verify adequate resolution of identified issues.

Specifically, the inspectors evaluated the selection and configuration of radiation detectors and sample lines for the noble gas effluent monitoring system by direct observation of the condenser exhaust radiation monitors (2-RE-90-119, 2-RE-90-255, and 2-RE-90-256) and main steam line radiation monitors (2-RE-90-421, 2-RE-90-422, 2-RE-90-423, and 2-RE-90-424). The inspectors assessed the range, sensitivity, and energy response of the installed instruments to ensure design requirements were adequately met. The inspectors observed the monitoring locations of the installed radiation equipment to verify that the as-installed configuration would meet the response requirements of the systems. The inspectors also reviewed PER 757975 to ensure that previously identified interferences with the installation and mounting of main steam line radiation monitor 2-RE-90-423 were properly addressed and corrective actions were adequately implemented.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

As described below, the NRC identified two violations where installation of main steam line radiation monitors did not meet applicable requirements.

b.1 Inadequate Design Drawing for Installation of Radiation Monitor

Introduction: The inspectors identified a SL IV NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the applicant's failure to provide a design drawing that assured that main steam line radiation monitor 2-RE-90-421 would be capable of performing its intended design basis function.

Description: During a walk down inspection of the south main steam valve room, the inspectors observed the completed installation of radiation monitor 2-RE-90-421. The inspectors determined that the as-installed configuration of the radiation monitor assembly resulted in the detector being shielded from the steam line for which it was designated to monitor for radioactive effluents.

The inspectors' review of WO 10661795 determined the document contained guidance for the installation of 2-RE-90-421. Specifically, drawing revision authorization (DRA) 52342-030 provided details for installation of the radiation monitor assembly. The inspectors' review of the DRA determined that the drawing incorrectly required the radiation monitor to be installed in a configuration where internal lead shielding contained in the detector enclosure shielded the detector from the designated steam line.

The inspectors determined that the applicant's failure to provide a design drawing that assured that main steam line radiation monitor 2-RE-90-421 would be capable of performing its intended design basis function was a performance deficiency. The inspectors determined the performance deficiency was more than minor, in accordance with Appendix C, Minor Violations and Findings, of IMC 2517, "Watts Bar Unit 2 Construction Inspection Program," dated June 6, 2013, because it represented an improper work practice that included failures to establish or implement adequate procedure functions. If left uncorrected, the installed configuration could adversely impact the function of components required for post-accident monitoring of effluent release paths. The finding has a cross-cutting aspect in the Documentation component of the Human Performance area, as defined in NRC IMC 0310, because design drawings were not accurate [H.7].

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in Section 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies, are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, on September 1, 2015, the design basis requirement to monitor effluent radioactivity in main steam lines was not correctly translated into engineering drawing DRA 52342-031 for radiation monitor 2-RE-90-421. As a result, the monitor

was installed in a configuration that prevented the radiation monitor from detecting radioactivity in its designated steam line.

This finding was determined to be a SL IV violation using Section 6.5 of the NRC Enforcement Policy because it represented a failure to meet a regulatory requirement, including one or more QA criteria that had more than minor safety significance. Because this was a SL IV violation and the issue was entered into the corrective action program as CR 1078327, this violation is treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. This violation is identified as NCV 05000391/2015608-04, "Design Basis Not Accurately Translated Into Design Drawing."

b.2 Failure to Install Radiation Monitors In Accordance With the Design Drawing

Introduction: The inspectors identified a SL IV NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for failure to install main steam line radiation monitors 2-RE-90-422 and 2-RE-90-423 according to their design drawing.

Description: During a walk down inspection of the north main steam valve room, the inspectors observed the completed installation of radiation monitors 2-RE-90-422 and 2-RE-90-423. The inspectors determined that the as-installed configurations of the radiation monitors resulted in the detectors being shielded from the steam lines for which they were designated to monitor for radioactive effluents.

The inspectors noted that WO 10661795, contained guidance for installation of 2-RE-90-422 and 2-RE-90-423. Detailed guidance for installation of the two radiation monitors was contained in drawing DRA 52342-031. The inspectors compared the drawing to the installed configurations and determined that radiation monitors 2-RE-90-422 and 2-RE-90-423 were not installed according to DRA 52342-031. The radiation monitors were actually oriented in a configuration where internal lead shielding contained in the detector enclosures shielded the detectors from the designated steam lines.

The inspectors determined that the applicant's failure to perform installations in accordance with design drawings in order to ensure that main steam line radiation monitors 2-RE-90-422 and 2-RE-90-423 would be capable of performing their intended design basis function was a performance deficiency. Inspectors determined the performance deficiency was more than minor, in accordance with Appendix C, Minor Violations and Findings, of IMC 2517, "Watts Bar Unit 2 Construction Inspection Program," dated June 6, 2013, because it represented an improper work practice that included failures to establish or implement adequate procedure functions. If left uncorrected, the installed configuration could adversely impact the function of components required for post-accident monitoring of effluent release paths. The inspectors reviewed this finding against cross-cutting area components as described in IMC 0310, "Components Within the Cross-Cutting Areas," and determined that no cross-cutting aspect applied.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, on September 1, 2015, the configuration of main steam line radiation monitors 2-RE-90-422 and 2-RE-90-423 was not in accordance with design drawings. As a result, the radiation monitors were oriented in configurations that prevented the detectors from monitoring radioactivity in their designated steam lines.

This finding was determined to be a SL IV violation using Section 6.5 of the NRC Enforcement Policy because it represented a failure to meet a regulatory requirement, including one or more QA criteria that had more than minor safety significance. Because this was a SL IV violation and the issue was entered into the corrective action program as CR 1078327, this violation is treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. This violation is identified as NCV 05000391/2015608-05, "Failure to Install Radiation Monitors In Accordance With the Design Drawing."

c. Conclusion

The inspectors identified two violations involving three instances where installation of main steam line radiation monitors did not meet applicable requirements. In addition, further inspection will be required to observe the as-built configurations of the main steam line monitoring system; and to evaluate procedures for maintenance calibrations, system testing, and operation.

OA.1.5 (Discussed) Three Mile Island Action Item II.F.1: Containment High Range Radiation Monitors (Inspection Procedure 35007)

a. Inspection Scope

Background: As part of the response to the TMI accident, Section II.F.1 of NUREG-0737 required applicants to install high range monitoring instrumentation with the capability to detect and measure the radiation level within containment during and following an accident. The previous inspection of this TMI action item was documented in IIR 05000391/2013607 (ADAMS Accession No. ML13273A512). That report identified that design output products for in-containment monitoring of radiation were consistent with the requirements of NUREG-0737.

Inspection Activities: The inspectors interviewed the responsible system engineer and reviewed equipment qualification installation requirements, setpoint and scaling documents, setpoint and accuracy calculations, calibration procedures, human factors analysis and annunciator alarm response instructions for the containment high-range radiation monitors. The review was performed to verify the as-installed configuration was in compliance with the requirements outlined in NUREG-0737.

The scope of the inspection included direct observation of the installed monitors and associated displays and controls in the main control room. The field inspection was performed to verify the locations in upper and lower containment were in accordance with the requirements of NUREG-0737. The detectors are subject to water intrusion and the field inspection also verified the EQ requirements for the containment junction box were implemented in accordance with the requirements outlined in EDCR 52338. This included direct observation of drilled weep holes on the bottom of the junction box, seals on the conduit entering the detector assembly, and installation of the wiring trough cover plate. The inspection of the displays and controls in the main control room was performed to verify the displays were in accordance with the specifications. The

inspectors verified that a human factors analysis of the displays and controls was performed and documented.

The inspectors reviewed the associated equipment qualification change supplement (EQCS) to ensure that the equipment qualification requirements were translated from the EDCR to the EQCS for later addition to the corresponding combined Unit 1 and 2 equipment qualification data packages.

The inspectors reviewed the surveillance instruction for the 18 month channel calibration to verify that the instruction included performance of an in-place source verification.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

The inspectors determined that the alarm set points for the containment high range monitors' control room annunciators had been established in the SSD and accurately translated to the annunciator alarm response procedure.

The inspectors determined that the equipment qualification information releases (EQIRs), which are used to document actual field installation of equipment, had not been issued.

The inspectors determined that detector locations in Unit 2 were consistent with those established for Unit 1 and the EQ requirements were implemented and documented adequately. The inspectors determined that procedures addressing the requirement to conduct in-place calibrations had been developed. The actual calibrations were still pending for completion at a future date. The inspector's determined that detector documentation specified operating range and energy response consistent with the requirements of NUREG-0737.

c. Conclusions

The inspectors determined that activities were in progress to address the TMI action item. Further inspections will be required to evaluate the adequacy of procedures for maintenance calibrations, system testing, and operation.

OA.1.6 (Closed) Three Mile Island Action Item II.E.1.2 and Supplemental Safety Evaluation Report, Appendix HH, Open Item 75: Auxiliary Feedwater System Initiation and Flow Indication (Inspection Procedures 92719 and 70438)

a. Inspection Scope

Background: NUREG-0737, TMI Action Item II.E.1.2, requires a timely initiation of the auxiliary feedwater (AFW) system, as well as a safety-grade AFW flow indication powered from the emergency buses. Section 7.8.2 of SSER 23 (ADAMS Accession No. ML11206A499) documents the staff's conclusion in regard to Item II.E.1.2 for WBN Unit 2. The SSER documented Appendix HH Open item 75 which states, "The NRC staff will

verify that the test procedures and qualification testing are completed in WBN Unit 2 before fuel load.”

Inspection Activities: The inspectors reviewed AFW system design documents and drawings, to verify:

- the design provides for automatic initiation of the AFW system;
- automatic initiation signals and circuits are designed so that a single failure will not result in the loss of AFW system function;
- testability is a feature of the design;
- initiating signals, circuits, motor-driven pumps, and valves are powered from safety-related sources;
- manual capability to initiate the AFW system from the control room is maintained; and
- AFW flow indication is powered from safety-related sources consistent with satisfying power diversity requirements;

The inspectors reviewed surveillance instructions for AFW flow indicators to verify the capability in the control room to ascertain the actual performance of the AFW system when it is called to perform its intended function. The inspectors verified flow indication during test witnessing of AFW preoperational testing.

In addition, the inspectors either witnessed or reviewed completed test procedures listed below to verify automatic initiation logic of the AFW system:

- 2-PTI-003B-04, Section 6.19 - MDFP 2A-A Breaker Control Logic and Functional Test
- 2-PTI-003B-04, Section 6.20 - MDFP 2B-B Breaker Control Logic and Functional Test
- 2-PTI-003B-04, Section 6.21 - AFW Auxiliary Relays Test
- 2-PTI-003B-04, Section 6.22 - 2-FCV-1-15 & 2-FCV-1-16, SG 1 & 4 Steam Supply to TDAFW Pump, Logic and Stroke Time Test
- 2-PTI-003B-04, Section 6.23 - 2-FCV-1-17, Steam Header to TDAFW Pump, Logic and Stroke Time Test
- 2-PTI-003B-04, Section 6.24 - 2-FCV-1-18, Steam Header to TDAFW Pump, Logic and Stroke Time Test
- 2-PTI-003B-04, Section 6.25 - 2-FCV-1-51, TDAFW Pump Trip & Throttle Valve, Logic Test
- 2-PTI-003B-04, Section 6.26 - 2-FCV-1-52, TDAFW Pump Governor Valve, Logic Test

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

Based on a review of the applicant's final closure package and the above activities, the inspectors determined that the applicant met the requirements for AFW automatic initiation and flow indication. TMI Action Item II.E.1.2 and SSER Appendix HH Item 75 are closed.

OA.1.7 (Closed) NRC Bulletin 80-04: Analysis of Pressurized Water Reactor Main Steam Line Rupture with Continued Feedwater Addition (Inspection Procedure 92717)

a. Inspection Scope

Background: NRC Bulletin (BL) 80-04 identified the susceptibility of increasing the postulated containment pressure following a main steam line break (MSLB) inside containment in which continued feedwater addition occurred from the AFW system at run out conditions. The BL was originally viewed as informational to Watts Bar since it was not operating and was not specifically listed in the BL enclosure as having to respond. TVA identified an analysis was needed to demonstrate the adequacy of containment for this susceptibility. Unit 1 actions were completed by TVA which involved the determination of temperature profiles inside and outside of containment following a MSLB. NRC IR 50-390/85-60 and 50-391/85-49 issued in December 1985 identified the subject BL to remain open for Unit 2 since it was unclear that the actions taken for Unit 1 would also be applicable to Unit 2. The MSLB analysis for Unit 2 was completed by Westinghouse in December 2009, and transmitted to TVA (LTR-CRA-11-196/WBT-D-3356) and was incorporated into the Unit 2 FSAR in Amendment 104, Section 6.2.1.3.10, "Steam Line Break Inside Containment." The analysis was re-performed by Westinghouse and transmitted to TVA (LTR-CRA-15-123) on September 1, 2015, incorporating the evaluation of a five-second delay for the closure of the motor-operated main feedwater isolation valves for Watts Bar Unit 2. The FSAR, Section 6.2.1.3.10, was updated with this revised information in Amendment 114. Previous inspection of BL 80-04 was performed in IIR 05000391/2011610 (ADAMS Accession No. ML12034A202), which reviewed the 2009 Westinghouse analysis.

Inspection Activities: The inspectors reviewed the letter transmitted to the applicant by Westinghouse dated September 1, 2015, to verify that changes to the analysis inputs were considered in the containment steam line break analysis. Additionally, the inspectors either witnessed or reviewed test results that support the assumptions in the FSAR to verify that they remained valid. Specifically, the inspectors performed a cursory review of valve stroke time data from 2-PTI-003A-01, "Feedwater Isolation Valves," Rev. 0, to verify that the main feedwater isolation valves and main feedwater regulating valves closed within the assumed time upon receipt of a feedwater isolation signal. The inspectors performed a cursory review of valve stroke time data from 2-PTI-001-01, "Main Steam Isolation Valves and Bypass Isolation Valves," Rev. 1, to verify that the main steam isolation valves closed within the assumed time upon receipt of a steam line isolation signal. The inspectors performed a cursory review of test results from 2-PTI-03B-05, "Auxiliary Feedwater System Dynamic Test," Rev. 1, to verify that testing demonstrated that the auxiliary feedwater pumps' flow controllers functioned to prevent pump runout.

b. Observations and Findings

No findings were identified.

c. Conclusions

Based on a review of the applicant's final closure package, the above inspection activities, and previous inspection results, the inspectors determined that the applicant has taken adequate actions in response to BL 80-04. BL 80-04 is closed.

OA.1.8 (Closed) Three Mile Island Action Item II.B.1: Reactor Coolant System Vent (Inspection Procedures 92717 and 35007)

a. Inspection Scope

Background: As a result of the accident at TMI Unit 2, the NRC created a number of action items designed to improve a plant's ability to minimize accident occurrence and accident consequences. These TMI action items were initially outlined in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident" (ADAMS Accession No. ML072470524) and later clarified in NUREG-0737, "Clarification of TMI Action Plan Requirements" (ADAMS Accession No. ML102560051). This item was previously inspected in IIR 05000391/2014614 (ADAMS Accession No. ML14363A315).

Inspection Activities: Based on the information provided in the background section, the objective of this inspection was to witness testing and review testing data to make a determination as to whether TVA had adequately addressed TMI Action Item II.B.1 for WBN Unit 2. The inspection focused on witnessing the performance of 2-PTI-068-09, "Reactor Vessel Head Vent System" and reviewing vibrational data obtained via 2-PTI-999-01, "Operational Vibration Testing." The inspectors also reviewed changes made to the WB2 FSAR (WBNP-114) Section 5.5.6.2 to ensure that it was consistent with the WB1 FSAR.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspectors reviewed the applicant's final closure report and various completed actions associated with TMI Action Item II.B.1 to verify the adequacy of the applicant's actions. The inspectors concluded that the applicant's efforts were sufficient to satisfy the intent of the respective TMI Action Item. TMI Action Item II.B.1 is considered closed.

OA.1.9 (Closed) Unresolved Item 87-19-06, Nonsafety-Related Cable in Close Proximity to Safety-Related Cable in Electrical Panels

a. Inspection Scope

Background: This unresolved item (URI) was initiated in 1987, when NRC inspectors observed both non-safety and safety-related Division B cables in physical contact with each other in junction box 2-JB-292-1973-B. Separately the applicant identified, in condition adverse to quality report (CAQR) WBP871231, that there were cables that violated the train separation criteria at the bottom of panel M15/2. PER 143957 was issued by the applicant to document and track resolution of these two individual examples of failure to meet train separation criteria. EDCR-2 55127 was issued to

address cable separation within panels and junction boxes. This document identified 18 individual panels/junction boxes with safety related cables and nonsafety-related cables or other safety-related cables from different trains in close proximity to each other requiring separation resolution. Previous inspection activities that addressed this item were documented in IIR 05000391/2010605 Section OA.1.15 (ADAMS Accession No. ML110410680), IIR 05000391/2014605 Section OA.1.1 (ADAMS Accession No. ML14226A049), and IIR 05000391/2015607 Section OA.1.13 (ADAMS Accession No. ML15273A452).

Inspection Activities: The conclusion from IIR 05000391/2015607 stated that the observations of completed work demonstrated that physical separation and electrical isolation between wiring systems covered under EDCR 55127 were adequately addressed. This inspection also satisfied the necessary inspection items needed to close this item.

b. Observations and Findings

No findings were identified.

c. Conclusions

The inspection activities discussed in the three inspection reports listed above demonstrated that the issues associated with URI 87-19-06 had been adequately resolved. URI 87-19-06 is closed.

OA.1.10 (Closed) Temporary Instruction 2515/152: Reactor Pressure Vessel Lower Head Penetration Nozzles and NRC Bulletin 2003-02: Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity (Temporary Instruction 2515/152)

a. Inspection Scope

Background: As part of the response to issues associated with degradation of the RPV upper head at the Davis-Besse Nuclear Power Station, the NRC issued Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002 (ADAMS Accession No. ML020770497), to request information about the condition and inspection of RPV upper heads as well as information about the licensee's boric acid corrosion control programs. The NRC subsequently issued Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," dated August 9, 2002 (ADAMS Accession No. ML022200494), to address staff concerns regarding the adequacy of visual examinations as a primary inspection method for the RPV upper head and RPV upper head penetrations.

The NRC staff summarized its review of the responses to Bulletin 2002-01 and the associated requests for additional information in Regulatory Issue Summary (RIS) 2003-13, "NRC Review of Responses to Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated July 29, 2003 (ADAMS Accession No. ML032100653). The NRC noted in RIS 2003-13 that most licensees did not perform inspections of Alloy 600/82/182 materials beyond those required by Section XI of the ASME Code to identify potential cracked and leaking

components. For the RPV lower head, the ASME Code specified that a visual examination, called a VT-2, be performed during system pressure testing. Many licensees performed the ASME Code required inspections without removing insulation and, therefore, most likely were not able to detect the amounts of through-wall leakage that would be expected from flaws due to primary water stress corrosion cracking or other potential cracking mechanisms. Degradation of RPV lower head penetration (LHP) nozzles and corrosion of the RPV lower head pose a safety concern because of the possibility of a nozzle ejection or loss-of-coolant accident if the conditions are not detected and repaired.

On August 21, 2003, the NRC issued Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," to:

- i. Advise pressurized water reactor (PWR) addressees that current methods of inspection of the RPV lower heads may need to be supplemented with additional measures (e.g., bare-metal visual inspections) to detect reactor coolant pressure boundary leakage;
- ii. Request PWR addressees to provide the NRC with information related to inspections that have been or will be performed to verify the integrity of the RPV lower head penetrations; and
- iii. Require PWR addressees to provide a written response to the NRC in accordance with the provisions of 10 CFR 50.54(f).

Inspection Activities: The inspectors observed the Watts Bar Unit 2 bare metal visual examination of the RPV lower head (directly and via remote camera) to verify its integrity and the absence of boric acid crystals, which may be evidence of a leak in the LHP nozzles. The inspectors also reviewed associated documentation, including the applicant's procedures, personnel and equipment qualifications, and final report. Specifically, the inspectors completed the following activities:

- a. The inspectors reviewed the qualifications, certifications, and training records of the inspection personnel and inspection techniques to verify the examiner was qualified as a VT-2 visual examiner, passed near and far distance acuity tests, as well as color discrimination and contrast tests, and completed a minimum of four additional hours of training in detection of borated water leakage from Alloy 600/82/182 components.
- b. The inspectors reviewed the preservice inspection (PSI) program plan and associated visual examination procedure to determine whether the documents provided adequate guidance and examination criteria to implement the applicant's examination plan.
- c. The inspectors conducted a performance-based inspection to verify that the applicant properly performed their procedure. The inspectors reviewed certificates of conformity and witnessed the camera's pre-resolution and post-resolution certifications to verify the camera could read lower case characters of height not greater than 0.105 inches at a distance of six inches as well as four feet away, representative of VT-3 sensitivity. The inspectors witnessed the method used to track the identification of the penetrations amongst the

procedure, video, and report to verify it was effective. The inspectors also observed the visual examination and reviewed the final report to confirm the examination (1) included 100% of the RPV lower head surface and 360 degrees of each penetration's annulus area (defined as the intersection between the RPV lower head and penetration, inclusive of 0.5 inches of adjacent RPV lower head base material) and (2) evaluated all relevant conditions including areas of corrosion, boric acid deposits, discoloration, and other evidence of nozzle leakage.

- d. The inspectors observed the visual examination of the RPV lower head both directly and remotely to determine whether the examination was conducted by properly qualified personnel in accordance with procedures and the applicant's approved quality assurance program. Their sample consisted of 13 LHP nozzles at different points distributed around the RPV lower head curvature, including the very center and outermost penetrations. The inspectors also reviewed a sample of records to verify they were prepared, evaluated, and maintained in accordance with applicable commitments and requirements.
- e. The inspectors also independently conducted a sample of six LHP nozzle examinations and assessed the condition of the RPV lower head to confirm all penetrations and surfaces were accessible and there was no evidence of boron leakage.

In accordance with NRC Bulletin 2003-02, the inspectors reviewed the applicant's PSI plan to evaluate the RPV lower head penetration inspection program that has been implemented for Watts Bar Unit 2. The inspectors also reviewed TVA's methodology for implementing the RPV lower head penetration inspection program that will be executed at Watts Bar Unit 2 during the next and subsequent refueling outages. The inspectors reviewed portions of the ASME Section XI in-service inspection (ISI) and NDE program plan for Unit 2 (Rev. 0), in conjunction with interviewing the Unit 2 PSI program engineer and coordinator, to determine whether the plan would meet the same requirements of the approved Unit 1 ASME Section XI ISI/NDE Program plan (Rev. 13) and NRC Bulletin 2003-02. The inspectors reviewed these plans to determine whether they included the extent of areas and penetrations to be inspected, inspection methods to be used, qualification standards for the inspection methods, the inspection documentation to be generated, and the basis for concluding the plant will satisfy applicable regulatory requirements related to the integrity of the RPV lower head penetrations.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

Based on the above activities, the inspectors concluded that Watts Bar Unit 2 is meeting the requirements of Bulletin 2003-02 using procedures, equipment, and personnel that have demonstrated to be effective in detecting signs of leakage from the RPV LHP

nozzles and the detection of RPV lower head degradation. Therefore, TI 2515/152 and Bulletin 2003-02 are considered closed.

OA.1.11 (Closed) Construction Deficiency Report 391/84-34: Spurious Valve Operation due to Fire Damage (Inspection Procedure 35007)

a. Inspection Scope

Background: The deficiency was initially reported to the NRC on July 30, 1984, in accordance with 10 CFR 50.55(e) as nonconformance report (NCR) 5760. The issue was documented as CDR WBRD 50-390/84-39 for Unit 1 and WBRD 50-391/84-34 for Unit 2. The CDR identified that the spurious actuation of an affected valve(s) during a fire could prevent the safe shutdown of the plant. This could result by spuriously closing essential safety-related flow paths or by spuriously diverting flow from an essential path to another non-essential system or component. This could adversely affect the safe operation of the plant; as stated in TVA's Final Report, "Spurious Valve Operation Due to Fire Damage," dated February 1, 1985 (ADAMS Accession No. ML082190672). During July 1984, TVA performed a 10 CFR Part 50, Appendix R review for WBN. During this review, several sets of redundant valves were observed to be capable of invalidating the safe shutdown analysis for WBN. This was due to potentially spurious valve operation as a result of fire damage to control circuitry. Three sets of valves were identified which affected the ERCW and the component cooling systems (CCS). TVA determined the cause of the deficiency to be a design oversight. This resulted from not having an established procedure for performing a fire protection analysis or for reviewing previously implemented designs to ensure compliance with 10 CFR Part 50, Appendix R.

To resolve the issue for Unit 1, TVA completed a review of all 10 CFR Part 50, Appendix R, shutdown paths in accordance with OE-SEP 84-09, "Safe Shutdown Analysis for Postulated Fire at Watts Bar Nuclear Plant," which was issued to prevent recurrence of this deficiency. OE-SEP 84-09 was replaced with Design Criteria WB-DC-40-51 "Fire Protection of Safe Shutdown Capability (Unit 1/Unit 2)." All valves capable of invalidating the Appendix R safe shutdown analysis due to spurious operation as a result of fire damage to control circuitry were identified and corrected via engineering change notices (ECNs) 5317, 5046, 5047, 5229, and 5338. These ECNs were developed to reroute control cables, re-pull portions of control cables, relocate control cables, wrap control cables, or remove power to prevent spurious valve operation as a result of fire damage to control circuitry. CDR 390/84-39 was closed for Unit 1 in NRC Inspection Report 50-390/85-09 (ADAMS Accession No. MO082190704).

For Unit 2, TVA initiated PER 172596 which included the commitment to evaluate existing cable routing for valves (capable of invalidating the Appendix R Safe Shutdown paths) and to identify all required control circuit wiring modifications required to prevent spurious valve operation due to fire damage and to perform the necessary modifications. PER 172596 also included the corrective actions that were identified in NCR 5760 (described above). Calculation EDQ00099920090012, "Unit 1 and 2 Appendix R Safe Shutdown Analysis," Rev. 2, and WB-DC-40-51, "Fire Protection of Safe Shutdown Capability (Unit 1/Unit 2)," Rev. 6, were issued to address historical design control deficiencies. TVA issued Appendix J of calculation EDQ00099920090012, which listed 46 required component modifications for Unit 2. The resolutions for the 46 components

were addressed in the following change documents: 52945, 53176, 53217, 53287, 53421, 53580, 53756, 54103, 54144, 54172, 54255, 54870, and 54912.

Inspection Activities: The inspectors reviewed Unit 2 modifications that were performed in accordance with EDCR 53217, 53287, and 54103 to verify that:

- cable re-routing, and other circuit modifications have been completed;
- Appendix R separation requirements were met for the circuits of concern; and
- the as-built configuration was consistent with the as-constructed Fire Protection Report (FPR).

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusion

The inspectors concluded that corrective actions completed or planned are adequate to resolve issues associated with Construction Deficiency Report 391/84-34, Spurious Valve Operation. No further inspection is required. CDR 391/84-34 is closed.

OA.1.12 (Closed) Construction Deficiency Report 391/84-35: Unacceptable Electrical Cable Separation (Inspection Procedure 35007)

a. Inspection Scope

Background: The deficiency was initially reported to the NRC on July 23, 1984, as NCR WBN EEB 8422 and 5761 in accordance with 10 CFR 50.55(e). The issue was documented as CDR WBRD 50-390/84-37 and WBRD 50-390/84-40 for Unit 1 and WBRD 50-391/84-35 for Unit 2.

The deficiency concerned unacceptable electrical cable separation conditions that involved Appendix R fire protection horizontal spatial requirements between redundant trains of safe shutdown equipment. The NCRs identify several instances where various Unit 1 and common cables did not meet Appendix R separation requirements. A single exposure fire which involves redundant cables or intervening combustibles could produce effects (e.g., smoke, heat, or ignition) which could adversely affect other redundant cables of a safe shutdown system. Both affected redundant cables could be adversely affected in the event of a fire from intervening combustibles. Thus, a single postulated fire could prevent the accomplishment of operations necessary for safe plant shutdown. This could adversely affect the safe operation of the plant. TVA determined the cause of the deficiency was a design oversight. This oversight was the lack of an established procedure for performing a fire protection analysis or for reviewing previously implemented designs to ensure compliance with 10 CFR Part 50, Appendix R.

For Unit 1, TVA performed additional reviews and analysis of all Appendix R requirements. TVA issued the Appendix R Safe Shutdown Analysis for Unit 1 in

Calculation WEBNOSG4031, "Equipment Required for Safe Shutdown PER 10CFR50 Appendix R." All equipment, circuits, and cables required for safe shutdown were identified along with any required separation between redundant trains for individual rooms and fire zones in the calculation. TVA issued ECNs 5338, 5229, and 5046 to reroute several cables to meet Appendix R requirements. In addition, TVA issued design criteria WB-DC-40-51, "Fire Protection of Safe Shutdown Capability (Unit1/Unit2)," and WB-DC-30-13, "10CFR, Appendix R, Type I, II, and III Circuits (Unit1/Unit2)," to address the design control corrective actions. The NRC closed CDRs 50-390/84-37 and 50-390/84-40 in Inspection Report 50-390/85-09 and 50-391/85-09 (ADAMS Accession No. ML082190704) to Inspection Follow-Up Item (IFI) 390/84-55-01. IFI 390/84-55-01 was closed in IR 50-390/85-29 and 50-391/85-25 (ADAMS Accession No. ML0821705903).

For Unit 2, TVA issued PER 172795 to address CDR 391/84-35. TVA combined the Unit 1 and Unit 2 Appendix R Safe Shutdown Analysis, and has issued the as-designed Calculation EDQ00099920090012, "Unit 1 and Unit 2 Appendix R Safe Shutdown analysis," Rev. 0. Appendix J of the calculation identified all actions required to have the 10 CFR Part 50, Appendix R, separation requirements for safe shutdown of Unit 1 and Unit 2 satisfied. TVA issued DCN 54912 and EDCRs 54633, 54636, 54637, 54639, 54640, 55231, 55233, and 53217 to resolve all actions as identified in Appendix J of calculation EDQ00099920090012.

Inspection Activities: The inspectors reviewed Unit 2 modifications that were performed in accordance with DCN 54912 and EDCRs 54633, 54636, 54637, 54640, 55231, 55233, and 53217 to verify that:

- cable re-routing, and other circuit modifications have been completed;
- Appendix R separation requirements were met for the circuits of concern; and
- the as-built configuration was consistent with the as-constructed FPR.

b. Observations and Findings

No findings were identified.

c. Conclusion

The inspectors concluded that corrective actions completed or planned are adequate to resolve issues associated with Construction Deficiency Report 391/84-35, Unacceptable Electrical Cable Separation. No further inspection is required. CDR 391/84-35 is closed.

OA.1.13 (Closed) Review of Actions Associated with EA-15-075 for Drilling Through Reinforcing Steel Without Obtaining Approval in Accordance with Project Procedures (Inspection Procedure 92702)

a. Inspection Scope

The inspectors reviewed the applicant's response to the SL IV Violation (EA-15-075) described in NRC IIR 05000391/2015604 (ADAMS Accession No. ML15181A446) and identified as violation (VIO) 05000391/2015604-02: Failure to Follow Anchor Bolt Installation Procedure. The SL IV Violation was associated with contract employees

willfully drilling through reinforcing steel of a steam generator housing without notifying site engineering or obtaining their approval in accordance with TVA procedure MAI-5.1B

The inspectors reviewed the applicant's apparent cause evaluation (ACE), associated PER, verified that all corrective actions to prevent recurrence (CAPR) were completed, and reviewed associated effectiveness reviews. Corrective actions that were verified included the revision of site and fleet procedures to either clarify or enhance the requirement of 10 CFR 50.9 and training for staff regarding 10 CFR 50.9. The applicant's ACE noted that reinforcing steel (rebar) had been cut or nicked at select locations during the installation of the anchor bolts without notifying engineering or obtaining an evaluation. Corrective actions were generated that included training on anchor bolt installation and revising procedures to require concrete to be scanned for rebar prior to drilling for anchor bolt installation. The inspectors reviewed the training and procedure revisions to verify if they were accomplished and adequate. The inspectors also reviewed the engineering evaluations performed to verify the structural adequacy of the structure containing the cut or nicked rebar. In addition, the inspectors sampled new work orders to verify that the requirement for scanning concrete structures for rebar prior to drilling for anchor bolt installation was incorporated.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

The applicant's causal analysis and corrective actions for EA-15-075 were adequate, fully implemented, and compliance with the associated regulatory requirements has been restored. VIO 05000391/2015604-02: Failure to Follow Anchor Bolt Installation Procedure, is closed.

OA.1.14 (Closed) Generic Letter 89-10: Safety-Related Motor Operated Valve Testing and Surveillance (Temporary Instruction 2515/109)

a. Inspection Scope

During an in-office review, the inspectors reviewed motor operated valve (MOV) calculations for the following GL 89-10 MOVs to determine if the applicant utilized an acceptable method, as per GL 89-10, to demonstrate the design basis of each MOV:

- 2-FCV-26-0243;
- 2-FCV-3-126A-B;
- 2-FCV-3-87A;
- 2-FCV-62-63A;
- 2-LCV-62-133B;
- 2-FCV-63-006;
- 2-FCV-63-007;
- 2-FCV-63-008;
- 2-FCV-63-011;

- 2-FCV-63-94B;
- 2-FCV-63-157B;
- 2-FCV-70-133A;
- 2-FCV-70-134B;
- 2-FCV-72-21B;
- 2-FCV-72-40A;
- 2-FCV-72-44A;
- 2-FCV-74-21B;
- 2-FCV-74-33A; and
- 2-FCV-74-08A.

The inspectors reviewed applicant activities in response to Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" and performed Part 3 inspections as documented in TI 2515/109. The inspectors performed these reviews to determine if the applicant had established and was implementing a program that will ensure the proper performance of MOVs in safety-related systems. Specifically, the inspectors determined if the applicant established an adequate design basis for each sampled MOV by reviewing the following assumptions used in the MOV calculations:

- valve factor (including area assumption);
- stem friction coefficient;
- load sensitive behavior (rate of loading);
- margins for stem lubrication degradation and spring pack relaxation;
- use of Limitorque, Kalsi Engineering Inc., or other sources for increasing thrust and torque allowable limits, as applicable;
- equipment error;
- degradation assumptions; and
- justification for grouping of MOVs for application of test data, performance characteristics, structural operating limits, and common-cause failure analyses.

The inspectors previously reviewed applicant actions to address GL 89-10; therefore, this inspection was a limited scope inspection of the requirements of this TI. Previous inspection activities for this TI as summarized below.

TI 2515/109 Section	Inspection Reports	ADAMS Accession Number
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Part 1:

04.01	• 05000391/2010604, Section OA.1.5 • 05000391/2011608, Section OA.1.11 • 05000391/2013610, Section OA.1.4	• ML103060240 • ML11311A082 • ML14049A158
04.02	• 05000391/2010604, Section OA.1.5 • 05000391/2011608, Section OA.1.11 • 05000391/2013610, Section OA.1.4	• ML103060240 • ML11311A082 • ML14049A158
04.03	• 05000391/2010604, Section OA.1.5 • 05000391/2011608, Section OA.1.11 • 05000391/2013610, Section OA.1.4	• ML103060240 • ML11311A082 • ML14049A158

Part 2:

04.04	• 05000391/2011608, Section OA.1.11 • 05000391/2013608, Section OA.1.2 • 05000391/2013610, Section OA.1.4 • 05000391/2014614, Section OA.1.3 • 05000391/2014615, Section OA.1.2 • 05000391/2015604, Section OA.1.3	• ML11311A082 • ML13316A776 • ML14049A158 • ML14363A315 • ML15044A424 • ML15181A446
04.05	• 05000391/2011608, Section OA.1.11 • 05000391/2013608, Section OA.1.2 • 05000391/2013610, Section OA.1.4 • 05000391/2014614, Section OA.1.3 • 05000391/2014615, Section OA.1.2 • 05000391/2015604, Section OA.1.3	• ML11311A082 • ML13316A776 • ML14049A158 • ML14363A315 • ML15044A424 • ML15181A446
04.06	• 05000391/2011608, Section OA.1.11 • 05000391/2013608, Section OA.1.2 • 05000391/2013610, Section OA.1.4 • 05000391/2014615, Section OA.1.2 • 05000391/2015604, Section OA.1.3	• ML11311A082 • ML13316A776 • ML14049A158 • ML15044A424 • ML15181A446
04.07	• 05000391/2011608, Section OA.1.11 • 05000391/2013608, Section OA.1.2 • 05000391/2013610, Section OA.1.4 • 05000391/2014607, Section OA.1.5 • 05000391/2014614, Section OA.1.3 • 05000391/2014615, Section OA.1.2 • 05000391/2015604, Section OA.1.3	• ML11311A082 • ML13316A776 • ML14049A158 • ML14274A076 • ML14363A315 • ML15044A424 • ML15181A446
04.08	• 05000391/2011608, Section OA.1.11 • 05000391/2013610, Section OA.1.4 • 05000391/2014615, Section OA.1.2 • 05000391/2015604, Section OA.1.3	• ML11311A082 • ML14049A158 • ML15044A424 • ML15181A446
04.09	• 05000391/2011608, Section OA.1.11 • 05000391/2013610, Section OA.1.4 • 05000391/2014615, Section OA.1.2 • 05000391/2015604, Section OA.1.3	• ML11311A082 • ML14049A158 • ML15044A424 • ML15181A446
04.10	• 05000391/2011608, Section OA.1.11 • 05000391/2013610, Section OA.1.4 • 05000391/2014615, Section OA.1.2 • 05000391/2015604, Section OA.1.3	• ML11311A082 • ML14049A158 • ML15044A424 • ML15181A446
04.11	• 05000391/2011608, Section OA.1.11 • 05000391/2013610, Section OA.1.4 • 05000391/2014614, Section OA.1.3 • 05000391/2014615, Section OA.1.2 • 05000391/2015604, Section OA.1.3	• ML11311A082 • ML14049A158 • ML14363A315 • ML15044A424 • ML15181A446

In accordance with requirements of TI 2515/109, Rev. 4, the inspectors evaluated and answered the following questions for Part 3:

- 04.12 Verify the scope of the licensee's GL 89-10 program and the status of each GL 89-10 MOV.

The inspectors reviewed calculation MDQ0029992009-0310, Rev. 9, "Generic Letter 89-10 MOV Population for Watts Bar (Unit 2)" and determined that the scope and status of valves in the applicant's GL 89-10 program was consistent with the requirements of GL 89-10.

- 04.13 Verify that the licensee has completed its design-basis review of GL 89-10 MOVs.

The inspectors reviewed calculation MDQ0029992009-0310, Rev. 9, to determine if the applicant completed the design basis review of all valves within the scope of GL 89-10. For each valve in the GL 89-10 program, the inspectors reviewed the valve factors, rate of loading, and stem factors as described in calculation MDQ0029992009-0310 and determined that the values used were reasonable. The inspectors reviewed the MOV thrust calculations for 19 GL 89-10 valves and concluded that the applicant adequately established the design basis for each valve.

- 04.14 Verify that the licensee has the established size and setting of GL 89-10 MOVs.

Based on this and previous MOV inspections, the inspectors concluded that the applicant properly established the size and settings of GL 89-10 MOVs. All MOVs within the scope of GL 89-10 have at least 5% margin. Previous inspections of static and dynamic testing verified proper limit and torque switch settings.

- 04.15 Verify that the licensee has completed its demonstration of design-basis capability of GL 89-10 MOVs.

The inspectors reviewed calculation MDQ0029992009-0310, Rev. 9, to determine if the applicant completed the design basis demonstration of all valves within the scope of GL 89-10. The inspectors reviewed the MOV thrust calculations for 19 GL 89-10 valves and concluded that the applicant adequately established the design basis for each valve. The inspectors determined that the methodology used to demonstrate the design basis for each valve was consistent with the requirements of GL 89-10.

- 04.16 Verify that the licensee has established plans for periodic verification of MOV design-basis capability.

The inspectors reviewed calculation MDQ0029992015000687, Rev. 1, "Watts Bar Nuclear Plant Unit 2 JOG MOV Periodic Verification Classification," and determined that the applicant established plans for the periodic verification of MOV design-basis capability. Subsequent to the issuance of GL 89-10, the NRC issued GL 96-05 which provided requirements to periodically verify the design basis capability of all GL 89-10 scoped MOVs. Inspection activities for GL 96-05 and the associated TI, TI 2515/140, provide more comprehensive review of the applicant's plans which supplants the necessity for additional inspection of the same area for GL 89-10.

- 04.17 Verify that the licensee has established an adequate process to analyze and trend MOV problems.

The applicant uses the corrective action program to analyze and trend MOV problems, which the inspectors concluded was an adequate process to perform this function.

- 04.18 Verify that the licensee has met its GL 89-10 schedule commitments.

All applicable and relevant schedule commitments have been met. The inspectors determined that the applicant has established an effective GL 89-10 program prior to unit startup and operation.

- 04.19 Verify that the licensee has implemented its quality assurance program with respect to GL 89-10 program.

The inspectors determined that the applicant's GL 89-10 program was being controlled in accordance with the requirements of its quality assurance program.

- 04.20 Verify that the licensee has resolved open, unresolved, and follow-up items from previous GL 89-10 inspections.

No open, unresolved, or follow-up items from previous GL 89-10 inspections exist.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings of significance were identified.

c. Conclusions

Based on this inspection and previous inspection activities documented in the inspection reports listed above, the inspectors concluded that the applicant's MOV program met the requirements of GL 89-10 and that the requirements of TI 2515/109 have been met for Watts Bar Unit 2. GL 89-10 and TI 2515/109 are closed.

OA.1.15 (Closed) Generic Letter 96-05: Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves (Temporary Instruction 2515/140)

a. Inspection Scope

During an on-site review, the inspectors reviewed applicant activities to periodically verify that MOV design-basis capability was consistent with the NRC regulations in 10 CFR 50.55a(b)(3)(ii) and the requirements of GL 96-05. To perform this review the inspectors sampled the following 33 valves that were within the applicant's GL 89-10 program scope:

- 2-FCV-070-0087-B
- 2-FCV-070-0089-B

- 2-FCV-070-0090-A
- 2-FCV-070-0092-A
- 2-FCV-070-0100-A
- 2-FCV-070-0133-A
- 2-FCV-070-0134-B
- 2-FCV-070-0140-B
- 2-FCV-070-0143-A
- 2-FCV-070-0153-B
- 2-FCV-070-0156-A
- 2-FCV-070-0183-A
- 2-FCV-070-0215-A
- 2-FCV-072-0002-B
- 2-FCV-072-0013-B
- 2-FCV-072-0021-B
- 2-FCV-072-0022-A
- 2-FCV-072-0034-A
- 2-FCV-072-0039-A
- 2-FCV-072-0040-A
- 2-FCV-072-0041-B
- 2-FCV-072-0044-A
- 2-FCV-072-0045-B
- 2-FCV-074-0001-A
- 2-FCV-074-0002-B
- 2-FCV-074-0003-A
- 2-FCV-074-0008-A
- 2-FCV-074-0009-B
- 2-FCV-074-0012-A
- 2-FCV-074-0021-B
- 2-FCV-074-0024-B
- 2-FCV-074-0033-A
- 2-FCV-074-0035-B

Because the applicant was a member of the Joint Owner's Group (JOG) and committed to the JOG program in the Update Final Safety Analysis Report (UFSAR), the inspectors reviewed the specific attributes of the applicant's JOG program including proper classification of the valves, service conditions, qualifying basis, and applied valve factor to:

- a. Verify the applicant was following the JOG program in classifying MOVs based on valve type, construction, service conditions, manufacturer, and their susceptibility to degradation;
- b. Verify the probabilistic risk assessment (PRA) and other justifications supported the assigned risk ranking and that that risk ranking along with the calculated thrust margin met the criteria for determining the test frequency for each MOV; and
- c. Determine whether the applicant completed the MOV classification process, documented the results, and whether the results were consistent with the JOG classification.

The inspectors previously reviewed applicant actions to address GL 96-05 and the requirements of this TI, which were documented in past inspection reports as summarized below.

TI 2515/140 Section	Inspection Reports	ADAMS Accession Number
03.01	• 05000391/2010604, Section OA.1.5 • 05000391/2011608, Section OA.1.11	• ML103060240 • ML11311A082
03.02	• 05000391/2010604, Section OA.1.5 • 05000391/2011608, Section OA.1.11	• ML103060240 • ML11311A082
03.03	• 05000391/2010604, Section OA.1.5 • 05000391/2011608, Section OA.1.11	• ML103060240 • ML11311A082

In accordance with requirements of TI 2515/140, the inspectors evaluated and answered the following questions for Part 3:

- 03.02 Verify the implementation of plans and commitments made as part of the completion of the MOV program in response to Generic Letter 89-10.

As described in Section OA.1.14, the inspectors verified that the applicant had developed and was implementing an adequate MOV program that met the requirements of GL 89-10 and commitments made in response to GL 89-10.

- 03.03 Determine whether the licensee has established and is implementing a program to provide continued assurance that the MOVs within the scope of GL 96-05 are capable of operating under design basis conditions. On the basis of a sample of MOVs, the inspector should evaluate the licensee's justification for the following aspects of its GL 96-05 program:

- a. Scope of the GL 96-05 program.

The inspectors reviewed calculation MDQ0029992015000687, Watts Bar Nuclear Plant JOG MOV Periodic Verification Classification, Rev. 1, and determined that the scope of valves in the GL 96-05 program was identical to the scope of valves in the GL 89-10 program. As described in Section OA.1.14, the inspectors verified that the scope of valves in the GL 89-10 program was adequate.

- b. Current design basis of the MOVs in the GL 96-05 program.

As described in Section OA.1.14, the inspectors concluded that the applicant adequately determined the current design basis for all valves in the GL 89-10 MOV program; therefore, the current design basis of the MOVs in the GL 96-05 program is also verified to be adequate.

- c. Degradation rate for the potential increase in the thrust or torque (as applicable) requirements to operate the valves.

All MOVs in the applicant's GL 96-05 program are classified as JOG category A and B valves. As such, these valves have been demonstrated by the JOG program to not be susceptible to degradation. Nevertheless,

the inspectors noted that the applicant included a degradation factor in the MOV thrust calculations to account for possible degradation. The inspectors determined that the degradation factor used in the calculations was appropriate.

- d. Degradation rate for the potential decrease in MOV actuator output under dynamic conditions.

The inspectors verified that the applicant adequately accounted for dynamic conditions in the MOV thrust calculations for all GL 89-10 (and GL 96-05) MOVs.

- e. Periodic test method to identify age-related degradation affecting the valve thrust or torque requirements and actuator output.

The inspectors verified that the applicant used industry-accepted MOV diagnostic measuring and test equipment and methodology to identify all degradation mechanisms that could affect valve thrust and/or torque requirements and actuator capability. The applicant used Crane Nuclear MOVATS ® equipment and software.

- f. Evaluation of test data to justify MOV test intervals.

The inspectors verified that the applicant appropriately used the methodology contained in the JOG program to determine the MOV test intervals. The inspectors reviewed the JOG classification, risk ranking, and thrust margin for a sample of 33 MOVs and determined that the applicant adequately determined the test frequency for each MOV.

- g. Periodic test interval that ensures continued MOV design-basis capability until next scheduled test.

The inspectors verified that the applicant appropriately used the methodology contained in the JOG program to determine the MOV test intervals. The inspectors reviewed the JOG classification, risk ranking, and thrust margin for a sample of 33 MOVs and concluded that the applicant adequately determined the test frequency for each MOV.

- h. The inspector should verify that all elements of the MOV program are encompassed by the quality assurance criteria of Appendix B to 10 CFR Part 50.

As described in Section OA.1.14, the inspectors verified that all aspects of the applicant's MOV program were encompassed by the quality assurance criteria of Appendix B to 10 CFR Part 50.

Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings were identified.

c. Conclusions

Based on this inspection and previous inspection activities documented in the inspection reports listed above, the inspectors concluded that the applicant's MOV program met the requirements of GL 96-05, and that the requirements of TI 2515/140 have been met for Watts Bar Unit 2. GL 96-05 and TI 2515/140 are closed.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

An exit meeting was conducted on October 15, 2015, to present inspection results to Ms. Cooper and other members of your staff. The inspectors identified that no proprietary information had been received during the inspection and none would be used in the inspection report. The applicant acknowledged the observations and provided no dissenting comments.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Applicant personnel

Y. Aboufaida, TVA – Systems Engineer
G. Arent, TVA – Licensing Manager
A. Bangalore, Bechtel - Electrical Engineer
R. Banks, WBN2 Outage Manager
P. Blakley, TVA Operations
F. Bodine – TVA Programs Oversight Engineer
G. Bonine, CILRT Test Director
R. Bowling, WBN2 SI Project Manager
E. Brumfield, Bechtel Project QA Manager
J. Calle, Dual Unit Transition Manager
K. Clark, Graftel Test Engineer
C. Cooke, Senior Program Maintenance OE/Procedures
C. Deblok, WBN2 Surveillance Program Coordinator
T. Detchmندی, EP Supervisor
B. Enis, TVA Engineering Oversight
M. Gowin – TVA Corporate Programs Manager
J. Grant, Start-Up Engineer Manager
H. Hill, Graftel Test Engineer
S. Hilmes, TVA - Electrical Engineer
G. Johnson, CILRT Test Director
Z. Keeton, WBN2 PM Coordinator
J. Kepler, Engineering Group Supervisor
B. Mahoney – Bechtel SIT Test Director
D. Myers, Senior QA Manager
J. Newton, TVA – Startup
J. O'Dell, TVA - Regulatory Compliance
C. Ottenfeld, System Area Turnover Support Operations Manager
J. Ortiz, TVA Programs Oversight Engineer
L. Peterson, WBN2 Site Support Manager
J. Perrel, Program Manager, EP Procedures
D. Phillips, Bechtel SIT Test Director
R. Phillips, Start-Up Manager
R. Poole, TVA – Maintenance
T. Powell, Dual Unit Master Schedule Coordinator
J. Proffitt, TVA – Licensing
T. Ramis, Bechtel SIT Test Director
G. Scott, TVA – Licensing
D. Shutt, TVA – Licensing
C. Six – TVA Mechanical Engineer Group Supervisor
M. Skaggs, TVA – Senior Vice President
K. Studder, Maintenance Director
J. Swanson, Startup Test Lead
T. Wallace, WBN2 Senior Manager Operations Unit 2 Construction
T. Washburn, TVA – Startup
R. Wigall, TVA Engineering Manager
T. Wilburn, Chemistry Project Supervisor
N. Welch, TVA - Preoperational Startup Manager

INSPECTION PROCEDURES USED

IP 35007	Quality Assurance Program Implementation During Construction and Pre-Construction Activities
IP 37002	Construction Refurbishment Process - Watts Bar Unit 2
IP 37301	Comparison of As-Built Plant to FSAR Description
IP 42450B	Operating Procedures
IP 50053	Reactor Vessel and Internals Work Observation
IP 50055	Reactor Vessel and Internals Record Review
IP 50073	Mechanical Components - Work Observation
IP 52053	Instrument Components and Systems - Work Observation
IP 53053	Containment Penetrations (Mechanical) Work Observation
IP 63050	Containment Structural Integrity Test
IP 70307	Containment Integrated Leak Rate Test Procedure Review
IP 70312	Preoperational Test Witnessing
IP 70313	Containment Integrated Leak Rate Test
IP 70315	Engineered Safety Features Test Preoperational Test Witnessing
IP 70348	Main Feedwater Control System Test - Preoperational Test Procedure Review
IP 70370	Testing Piping Support and Restraint Systems
IP 70400	Preoperational Test Results Evaluation
IP 70434B	Engineered Safety Features Act. Sys. Test - Preoperational Test Witnessing
IP 70436	Residual / Decay Heat Removal System Test - Preoperational Test Witnessing
IP 70438	Auxiliary Feedwater System Test - Preoperational Test Witnessing
IP 71111.07	Heat Sink Performance
IP 71301	Technical Specification Review
IP 71302	Preoperational Test Program Implementation Verification
IP 73053	Preservice Inspections – Observation of Work and Work Activities
IP 83526	Control of Radioactive Materials and Contamination, Surveys, and Monitoring (Preoperational And Supplemental)
IP 84523	Liquids and Liquid Wastes (Preoperational and Supplemental)
IP 84524	Gaseous Waste System (Preoperational And Supplemental)
IP 92701	Followup
IP 92702	Followup on Corrective Actions for Violations and Deviations
IP 92717	IE Bulletins for Information and IE Information Notice Followup
IP 92719	Safety Evaluation Report (SER) Review and Followup
TI 2512/015	Inspection of Watts Bar Nuclear Plant Employee Concerns Program
TI 2512/041	Inspection of Radiation Monitoring System Special Program
TI 2515/109	Inspection Requirements for Generic Letter 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance
TI 2515/114	Inspection Requirements for Generic Letter 89-04, Acceptable Inservice Testing Programs
TI 2515/140	Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves (GL 96-05)
TI 2515/152	Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02)

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000391/2015608-01	NCV	Failure to Perform Stopwatch Calibrations on a Specified Interval (Section P.1.1)
05000391/2015608-02	NCV	Preconditioning Identified in Surveillance Instructions (Section P.1.2)
05000391/2015608-03	NCV	Failure to provide adequate instructions for plant modifications (Section P.1.11)
05000391/2015608-04	NCV	Design Basis Not Accurately Translated Into Design Drawing (Section OA.1.4)
05000391/2015608-05	NCV	Failure to Install Radiation Monitors In Accordance With the Design Drawing (Section OA.1.4)

Closed

50053	IP	Reactor Vessel and Internals Work Observation (Section C.1.2)
50055	IP	Reactor Vessel and Internals Record Review (Section C.1.3)
53053	IP	Containment Penetrations (Mechanical) - Work Observation (Section C.1.5)
71111.07	IP	Heat Sink Performance, Subissue: Essential Raw Cooling Water Testing and Supplemental Safety Evaluation Report (Section C.1.7)
Item 90	SSER Appendix HH	Emergency Raw Cooling Water Pumps Meeting General Design Criteria 5 Requirements for Two-Unit Operation (Section C.1.7)
88-11	BL	Pressurizer Surge Line Thermal Stratification (Section C.1.8)
37301	IP	Comparison of As-Built Plant to FSAR Description (Section P.1.2)
42450B	IP	Operating Procedures (Section O.1.1)
71301	IP	Technical Specifications Review (Section O.1.2)

II.E.1.2	TMI Action Item	Auxiliary Feedwater System Initiation and Flow Indication (Section OA.1.6)
Item 75	SSER Appendix HH	Auxiliary Feedwater System Initiation and Flow Indication (Section OA.1.6)
80-04	BL	Analysis of Pressurized Water Reactor Main Steam Line Rupture with Continued Feedwater Addition (Section OA.1.7)
II.B.1	TMI Action Item	Reactor Coolant System Vent (Section OA.1.8)
391/87-19-06	URI	Nonsafety-Related Cable in Close Proximity to Safety-Related Cable in Electrical Panels (Section OA.1.8)
2515/152	TI	Reactor Pressure Vessel Lower Head Penetration Nozzles (Section OA.1.10)
2003-02	BL	Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity (Section OA.1.10)
391/84-34	CDR	Spurious Valve Operation due to Fire Damage (Section OA.1.11)
391/84-35	CDR	Unacceptable Electrical Cable Separation (Section OA.1.12)
05000391/2015604-02	VIO	Failure to Follow Anchor Bolt Installation Procedure (Section OA.1.13)
89-10	GL	Safety-Related Motor Operated Valve Testing and Surveillance (Section OA.1.14)
2515/109	TI	Safety-Related Motor Operated Valve Testing and Surveillance (Section OA.1.14)
96-05	GL	Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves (Section OA.1.15)
2515/140	TI	Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves (Section OA.1.15)
<u>Discussed</u>		
83526	IP	Control of Radioactive Materials and Contamination, Surveys, and Monitoring (Section P.1.15)
84523	IP	Liquids and Liquid Wastes (Section P.1.16)

84524	IP	Gaseous Waste System (Section P.1.17)
89-04	GL	Guidance on Developing Acceptable In-Service Testing Programs (Section OA.1.1)
2515/114	TI	Inspection Requirements for Generic Letter 89-04, Acceptable In-Service Testing Programs (Section OA.1.1)
52053	IP	Instrument Components and Systems – Work Observation (Section OA.1.2)
2512/041	TI	Radiation Monitoring System Special Program (Section OA.1.3)
II.F.1	TMI Action Item	Noble Gas Effluent Monitors (Section OA.1.4)
II.F.1	TMI Action Item	Containment High Radiation Monitors (Section OA.1.5)

LIST OF DOCUMENTS REVIEWED

II. MANAGEMENT OVERSIGHT AND CONTROLS

C.1 Construction Activities

C.1.3 Reactor Vessel and Internals Record Review

WO 116225719, Disassemble/Reassemble the U2 Reactor Pressure Vessel IAW 2-MI-68.001

WO 117001713, Remove and Install Lower Internals per 2-MI-058.003

2-MI-68.003, Removal and Replacement of the Unit 2 Reactor Vessel Lower Internals, Rev. 0

C.1.6 Mechanical Components – Work Observation and Construction Refurbishment Process

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C.1.7 Inspection Procedure 71111.07, Heat Sink Performance, Subissue: Emergency Raw Cooling Water Testing; and Supplemental Safety Evaluation Report, Appendix HH, Open Item 90: Emergency Raw Cooling Water Pumps Meeting General Design Criteria 5 Requirements for Two-Unit Operation

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2-PTI-067-02-A, Test Summary Report for 2-PTI-067-020A, ERCW System Flow Balance-Train “A”, Rev. 4

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C.1.8 Testing Piping Support and Restraint Systems, and Bulletin 88-11: Pressurizer Surge Line Thermal Stratification

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2-PTI-999-02, “Thermal Expansion,” Rev 1

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1-47W845-2, Watts bar Nuclear Plant, TVA, Mechanical Flow Diagram Essential Raw Water System, Rev. 92

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2-PTI-068-03, Pressurizer Heater and Spray Control, Rev. 0

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 2-SI-3-901-A, "Motor Driven Auxiliary Feedwater 2A-A Quarterly Performance Test", Rev 0002
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CR 1078788

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Lesson Plan 3-OT-SYS090A, Radiation Monitoring; Rev. 19

P.1.3 Preoperational Test Procedure Review

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 1078710
 1080398
 1081214

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 2-47W801-2, Main & Reheat Flow Diagram, Rev. 29
 2-47W803-1, Feedwater Flow Diagram, Rev. 32
 2-47W803-2, Auxiliary Feedwater Flow Diagram, Rev. 32
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 2-47W809-7, Flood Mode Boration Makeup System Flow Diagram, Rev. 11
 2-47W810-1, Residual Heat Removal System Flow Diagram, Rev. 22
 2-47W811-1, Safety Injection System Flow Diagram, Rev. 45
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 2-47W813-1, Reactor Coolant System Flow Diagram, Rev. 34
 2-47W814-2, Ice Condenser Flow Diagram, Rev. 22

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 2-47W830-1, Waste Disposal System Mechanical Flow Diagram, Rev. 21
 2-47W830-6, Waste Disposal System Mechanical Flow Diagram, Rev. 23
 2-47W845-3, Raw Cooling Water Mechanical Flow Diagram, Rev. 19
 2-47W846-2, Control and Service Air System Flow Diagram, Rev. 5
 2-47W848-1, Control Air System Flow Diagram, Rev. 15
 2-47W848-5, Control Air System Flow Diagram, Rev. 14
 2-47W848-9, Control Air System Flow Diagram, Rev. 18
 2-47W848-10, Control Air System Flow Diagram, Rev. 18
 2-47W850-9, Fire Protection Flow Diagram, Rev. 19
 2-47W851-1, Floor & Equipment Drain Mechanical Flow Diagram, Rev. 16
 2-47W855-1, Fuel Pool Cooling and Cleaning System Flow Diagram, Rev. 11
 2-47W856-1, Demineralized Water and Cask Decon System Flow Diagram, Rev. 0
 2-47W859-3, Component Cooling System Mechanical Flow Diagram, Rev. 31
 2-47W865-5, Air Conditioning Chiller Water Flow Diagram, Rev. 19
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112183491
 115689350
 116347436
 116348205
 116585126
 116734468

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TVA Letter No. TVWES-0339, Document Submittal- LOCA Mass and Energy Release Input Assumptions, dated May 24, 2004

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Instruments 49727 - 49735, dated 6/23/15

Instruments 49816 - 49821, dated 6/22/15

Instrument 49863, dated 6/23/15

Instruments 49622 - 49637, dated 7/13/15

Instruments 49905 - 49906, dated 6/30/15

Instruments 50027 - 50028, dated 6/23/15

Instruments 50030 - 50032, dated 6/12/15

TVA Calibration Certificates:

Instrument 18298, dated 12/18/09

Instrument 21068, dated 2/16/10

Instrument 24810, dated 6/10/10

Instrument 26969, dated 8/05/10

Instrument 34509, dated 12/17/10

Instrument 36364, dated 1/31/11

Instrument 39580, dated 3/29/11

Instrument 44942, dated 8/05/11

Instrument 53912, dated 1/27/12

Instrument 62607, dated 7/26/12

Instrument 70287, dated 12/21/12

Instrument 70622, dated 3/15/13

Instrument 81506, dated 8/20/13

Instrument 87421, dated 12/21/13

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P.1.4 Preoperational Test Witnessing

NC-PI-98-03 Foxboro I/A Network Storm Test, Rev 0

P.1.5 Preoperational Test Witnessing

Procedures

2-PTI-067-04, "Essential Raw Cooling Water (ERCW) Dual Unit Shutdown Flow Test," Rev. 0

Work Orders

116923934, Collect CCS Train B Flow Data With Both CCS Pumps In Service, Completed 8/27/15

116364053, 0-SI-70-902-S Component Cooling System Pump C-S Quarterly Performance Test, completed 5/20/15

116347481, 1-SI-70-901-B Component Cooling System Pump 1B-B Quarterly Performance Test, completed 4/24/15

116364179, 0-SI-70-903-S Component Cooling System Pump C-S Comprehensive Pump Test, completed 5/20/15

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Design Change Notice

DCN 35387A, CCS Pumps' Hydraulic Performance Did Not Meet Or Exceed Design Curve,
5/4/1995

Corrective Action Documents

PER 848656

Corrective Action Documents Generated As A Result Of Inspection

CR 1085478

P.1.8 Preoperational Test WitnessingCondition Reports Written During Testing

1081214

1073506

1078677

1078710

1079772

1080398

1080605

Condition Reports Reviewed

0145118

1062157

Procedures

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2-SI-0-703, Containment Integrated Leak Rate Test, Rev. 2, OTO-1, and OTO-2

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Power Reactors

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1995

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Watts Bar Unit 1, Technical Specifications, 5.7.2.19, Containment Leakage Rate Testing
Program, Amendment 5, 63

Watts Bar Unit 2 Technical Specifications (Developmental)

Watts Bar Unit 2 Technical Specifications Bases Document (Developmental)

Work Orders

010951323

115689350

116347436

116348205
 116585126
 116612462
 116687422
 116869301
 116903360
 116921589
 116946422
 116957972

Other Documents

72-4334 Contract, CBI Overload Pressure Test and Leakage Rate Test Report, Watts Bar Nuclear Plant Containment Vessel Unit No. 2, Watts Bar Tennessee for TVA
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 ANSI/ANS-56.8-1987, Containment System Leakage Testing Requirements
 ANSI/ANS-56.8-1994, Containment System Leakage Testing Requirements
 ASME Section III, ASME Boiler and Pressure Vessel Code, Winter 1971 Addenda and code cases 1431, 1517, 1529, 1493 and 1768.
 ASME Section XI, Subsection IWE and IWL, 2001 Edition through 2003 Addenda
 Commercial Dedication Document, Integrated Leak Rate Test System (ILRT), Rev. 0.2, dated 7/31/97
 Chicago Bridge & Iron Company Letter to TVA, Mr. R.G. Domer, TVA Reference No.: 73061-75320 Containment Vessels for Watts Bar Nuclear Plant: Units I and II CBI Contracts 72-4333/34 Communication No. 1.1.E-217, dated 8/25/1978
 TVA Letter No. TVWES-0339, Document Submittal- LOCA Mass and Energy Release Input Assumptions, dated May 24, 2004

Graftel Calibration Certificates:

Certificates 48910 - 48964, dated 6/12/15
 Certificates 49727 - 49735, dated 6/23/15
 Certificates 49816 - 49821, dated 6/22/15
 Certificate 49863, dated 6/23/15
 Certificates 49622 - 49637, dated 7/13/15
 Certificates 49905 - 49906, dated 6/30/15
 Certificates 50027 - 50028, dated 6/23/15
 Certificates 50030 - 50032, dated 6/12/15

TVA Calibration Certificates:

Certificate 18298, dated 12/18/09
 Certificate 21068, dated 2/16/10
 Certificate 24810, dated 6/10/10
 Certificate 26969, dated 8/05/10
 Certificate 34509, dated 12/17/10
 Certificate 36364, dated 1/31/11
 Certificate 39580, dated 3/29/11
 Certificate 44942, dated 8/05/11
 Certificate 53912, dated 1/27/12
 Certificate 62607, dated 7/26/12
 Certificate 70287, dated 12/21/12
 Certificate 70622, dated 3/15/13
 Certificate 81506, dated 8/20/13
 Certificate 87421, dated 12/21/13
 Certificate 95871, dated 6/24/14
 Certificate 104531, dated 12/18/14
 Certificate 112746, dated 6/17/15

P.1.10 Preoperational Test Witnessing

Work Order 115448018, "Perform PTI 2-099-04," dated 8/10/2015

TDN 15-1482, dated 8/11/2015, "Internal wiring found landed on metal instead of plastic"

TDN 15-1483, dated 8/11/2015, "TI 12-08 tags missing – Tags have wrong terminal numbers"

TDN 15-1487, dated 8/12/2015, "Terminal points found landed with plastic instead of hard landed in accordance with PTI"

P.1.12 Preoperational Test Results EvaluationProcedures

2-PTI-067-04, "Essential Raw Cooling Water (ERCW) Dual Unit Shutdown Flow Test," Rev. 0

Work Orders

116923934, Collect CCS Train B Flow Data With Both CCS Pumps In Service, Completed 8/27/15

116364053, 0-SI-70-902-S Component Cooling System Pump C-S Quarterly Performance Test, completed 5/20/15

116347481, 1-SI-70-901-B Component Cooling System Pump 1B-B Quarterly Performance Test, completed 4/24/15

116364179, 0-SI-70-903-S Component Cooling System Pump C-S Comprehensive Pump Test, completed 5/20/15

Miscellaneous

Test Summary Report for 2-PTI-067-04, Essential Raw Cooling Water (ERCW) Dual Unit Shutdown Flow Test, Rev. 0, 9/16/15

Design Change Notice

DCN 35387A, CCS Pumps' Hydraulic Performance Did Not Meet Or Exceed Design Curve, 5/4/1995

Corrective Action Documents

PER 848656

Corrective Action Documents Generated As A Result Of Inspection

CR 1085478

P.1.13 Preoperational Test Results EvaluationProcedures

SMP-8.0, Watts Bar Nuclear Plant Unit 2 Administration of Preoperational Test Instructions, Rev. 0011

SMP-3.0, Watts Bar Nuclear Plant Unit 2 Joint Test Group Charter, Rev. 0005

Corrective Action Documents

PER 954059

PER 956284

PER 907064

PER989830

CR 1029402

CR 1039165

CR 1006450

CR 1052304

CR 1068680

Work Orders

116880310

116672726

116932360

116931974

117072895

117072892

117164177

117164195

Miscellaneous

Individual Test Results Package

White Paper for CCS Flow Balance Results Train 2A

Joint Task Group Meeting Minutes dated March 26, 2015

Joint Task Group Meeting Minutes dated May 25, 2015

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Westinghouse Letter LTR-SEE-I-14-67

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P.1.14 Preoperational Test Results Evaluation

SMP-8.0, Watts Bar Nuclear Plant Unit 2 Administration of Preoperational Test Instructions, Rev. 0011

SMP-9.0, Watts Bar Nuclear Plant Unit 2 Conduct of Test, Rev. 0006

2-PTI-099-05, Overpower Delta-T and Overtemperature Delta-T Turbine Runback Test Summary Report, Rev.1

P.1.15 Control of Radioactive Materials and Contamination, Surveys, and Monitoring (Preoperational and Supplemental)

WBN-DCD-40-24, Radiation Monitoring (Unit 1/ Unit 2), Rev. 26

P.1.16 Liquids and Liquid Wastes (Preoperational and Supplemental)

Offsite Dose Calculation Manual, Rev. 25

WBN-DCD-40-24, Radiation Monitoring (Unit 1/ Unit 2), Rev. 26

TI-18, Calculation Methods for Effluent Monitors, Rev. 29

2-PTI-077-01, Liquid Waste Processing System, Rev. 1

2-PTI-077-03, Liquid Waste Collection, Rev. 1

SE SSD 2-R-90-120, Setpoint and Scaling Document 2-R-90-120, 7/22/15

2-PTI-090-05, Steam Generator Blowdown Liquid Effluent Radiation Monitors, Test Summary Report, 8/20/15

P.1.17 Gaseous Waste System (Pre-Operational and Supplemental)

DCN 54207-A, [Installation of Additional Flow Elements in Ductwork Feeding the Shield Building Exhaust Vent (SBEV)], 6/30/10

2-PTI-077-02, Gaseous Waste Disposal System Preoperational Test Instruction, Rev. 0

SSD-2-R-90-131, NE Setpoint and Scaling Document, Rev.1

SSD-2-R-90-130, NE Setpoint and Scaling Document, Rev.1

SSD-2-R-90-112A, B, C, NE Setpoint and Scaling Document, Rev.1

SSD-2-LPR-90-106, Setpoint and Scaling Document, Unit 2 Containment Lower Compartment Atmosphere Monitor, Rev. 1

SSD-2-LPR-90-112, Setpoint and Scaling Document, Unit 2 Containment Upper Compartment Atmosphere Monitor, Rev. 1

SSD-2-LPR-90-130, Setpoint and Scaling Document, Unit 2 Containment Purge Air Exhaust Radiation Monitor (Train A), Rev. 0

SSD-2-LPR-90-131, Setpoint and Scaling Document, Unit 2 Containment Purge Air Exhaust Radiation Monitor (Train B), Rev. 0

III. OPERATIONAL READINESS ACTIVITIES**O.1 Operations****O.1.1 Operating Procedures**Procedures

1-SI-0-2-00 (Rev 0080), "Shift and Daily Surveillance Log Master"

2-SI-0-2-00 (Rev 0003), "Shift and Daily Surveillance Log Master"

1-SI-0-2A-01 (Rev 0048), "1900-0700, Shift and Daily Surveillance Log Mode One"
 2-SI-0-2A-01 (Rev 0001), "1900-0700, Shift and Daily Surveillance Log Mode One"
 1-SI-0-2A-03 (Rev 0036), "1900-0700, Shift and Daily Surveillance Log Mode Three"
 2-SI-0-2A-03 (Rev 0000), "1900-0700, Shift and Daily Surveillance Log Mode Three"
 1-SI-0-2B-06 (Rev 0037), "0700-1900, Shift and Daily Surveillance Log Mode Six"
 2-SI-0-2B-06 (Rev 0001), "0700-1900, Shift and Daily Surveillance Log Mode Six"
 1-SI-0-4 (Rev 0032), "Monthly Surveillances"
 2-SI-0-4 (Rev 0002), "Monthly Surveillances"
 1-SI-92-1 (Rev 0026), "NIS Daily Comparison"
 2-SI-92-1 (Rev 0001), "NIS Daily Comparison"
 1-SI-262-1-A (Rev 0010), "Train A Auto Start Lockout Test"
 2-SI-262-1-A (Rev 0012), "Unit 2 Train A Auto Start Lockout Test"
 1-SI-262-1-B (Rev 0012), "Train B Auto Start Lockout Test"
 2-SI-262-1-B (Rev 0012), "Unit 2 Train B Auto Start Lockout Test"
 1-SI-268-1-A (Rev 0004), "92 Day Permanent Hydrogen Mitigation System Train A Igniter Availability Test"
 2-SI-268-1-A (Rev 0000), "92 Day Permanent Hydrogen Mitigation System Train A Igniter Availability Test"
 1-SI-268-1-B (Rev 0004), "92 Day Permanent Hydrogen Mitigation System Train B Igniter Availability Test"
 2-SI-268-1-B (Rev 0000), "92 Day Permanent Hydrogen Mitigation System Train B Igniter Availability Test"
 1-SI-268-2-A (Rev 0004), "18 Month Permanent Hydrogen Mitigation Systems Train A Operability Test"
 2-SI-268-2-A (Rev 0000), "18 Month Permanent Hydrogen Mitigation Systems Train A Operability Test"
 1-SI-268-2-B (Rev 0004), "18 Month Permanent Hydrogen Mitigation Systems Train B Operability Test"
 2-SI-268-2-B (Rev 0000), "18 Month Permanent Hydrogen Mitigation Systems Train B Operability Test"

Miscellaneous

ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants"
 ANSI N18.7-1976/ANS 3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"
 RG 1.33-1978, Revision 2, "Quality Assurance Program Requirements (Operation)"
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O.1.2 Technical Specifications Review

Procedures

2-SI-3-10, 18 Month Channel Calibration of Steam Generator 4 Narrow Range Level Channel II Loop 2-LPL-3-106 (L-549)
 2-SI-3-1, 18 Month Channel Calibration of Steam Generator 1 Narrow Range Level Channel II Loop 2-LPL-3-38 (L-519)
 2-SI-68-80-A, Channel Calibration of Train A Common Q Post Accident Monitoring System
 2-SI-0-4, Monthly Surveillances, Appendix D, Remote Shutdown and PAM Channel Check Data
 0-SI-65-6-B, Emergency Gas Treatment System Train B 10-Hour Operation
 0-SI-30-7-A, Auxiliary Building Gas Treatment System Pressure Test Train A
 0-82-5, 18 Month Loss Of Offsite Power With Safety Injection Test - DG 2A-A
 0-82-6, 18 Month Loss Of Offsite Power With Safety Injection Test - DG 2B-B

IV. OTHER ACTIVITIES

OA 1.1 Generic Letter 89-04: Guidance on Developing Acceptable In-Service Testing Programs; Temporary Instruction 2515/114: Inspection Requirements for Generic Letter 89-04, Acceptable In-Service Testing Programs

Procedures

2-SI-63-916-A, Safety Injection Pump 2A-A Preservice Pump Test, Rev. 2

2-SI-63-916-B, Safety Injection Pump 2A-A Preservice Pump Test, Rev. 2

Calculations

WBN-WBT-D-4376, Watts Bar Unit 2 ECCS Analysis Report, Rev. 2

OA.1.3 Radiation Monitoring System Special Program

Miscellaneous

CR 0143931, Design Input Requirements not Established for the Radiation Monitoring System, dated 10/26/1987

CR 0144996, Output Signal Transients are occurring, dated 5/2/1988

CR 0144193, Failures due to Inductive Kickback, dated 4/25/1991

CR 1077639, (NRC Identified) Detector Cable does not have a support, dated 9/1/2015

CR 1077643, (NRC Identified) 2-RE-90-59 detector cable is being supported by a tie-wrap

CR 1077645, (NRC Identified) Removal of power causes computer for 2-RM-90-120, -121 to lose its database, dated 9/22/2015

CR 1078327, (NRC Identified) WBN Unit 2 steam line rad monitors are not installed correctly, dated 9/3/2015

EDCR 52340, Rev. A, Modifications to RMS Loops in the Turbine Building

EDCR 52341, Rev. A, Modifications to RMS Loops in the Auxiliary Building

General Atomics Drawing 04024120, Rev. F, Customer Connection Diagram, High Range Area Monitor System

WBN-DCD-30-7, Rev. 27, Design Criteria - Post Accident Monitoring Instrumentation (Unit 1/Unit 2)

WBN-DCD-40-24, Rev. 25, Design Criteria - Radiation Monitoring (Unit 1/Unit 2)

OA.1.4 TMI Action II.F.1, Noble Gas Effluent Monitors

Miscellaneous

WB-DC-40-24, RMS Design Criteria

PER 757975, (NRC Identified) Main Steam Line Radiation Detector Mounting Plates, dated 09/14/2013

EDCR-2 52342, Revision A, dated 2/4/2010

DRA 52342-030, Revision 3, dated 7/7/2015

DRA 52342-031, Revision 2, dated 12/3/2010

Drawing 04033100, Revision F, dated 8/11/2011

TDN 15-1585, dated 9/2/2015

WO 10661795, Revision 0, dated 5/13/2015

OA.1.5 TMI Action II.F.1, Containment High Range Radiation Monitors

Miscellaneous

ARI-265-268, Annunciator Response Instruction Post Accident Radiation, dated 12/18/2013

2-SI-90-1, 18-Month Channel Calibration Containment Upper Compartment High Range Post Accident Area Radiation Monitor Loop 2-LPR-90-271-A, dated 7/10/2015

2502-011-V1A-HARA-00124-007, Qualification Basis for 2-RE-271,-272,-273,-274, dated February 2011

EDCR 52338, Modify Radiation Monitoring Loops in the Unit 2 Reactor Building for 2-RE-90-271A, 272B, 273A and 274B, dated 4/23/2010

EQCS No. WBNEQ-IRE-001-52338, Environmental Qualification Package General Atomic Rd-23 Radiation Monitors, dated 7/9/2012

SSD-2-R-90-271A, Setpoint and Scaling Document, dated 8/25/15

WBNTSR-027, Setpoint for the Containment High-Range Monitors, revision 6, dated 6/24/09

Work Order 116930593, Implement FCR 65637-A to Fabricate and Install Covers for the Radiation Monitors, dated 8/12/15

OA.1.6 Three Mile Island Action Item II.E.1.2 and Supplemental Safety Evaluation Report, Appendix HH, Open Item 75: Auxiliary Feedwater System Initiation and Flow Indication Procedures

2-PTI-03B-04; Auxiliary Feedwater Pumps & Valves Logic Test; Rev. 1

Drawings

2-47W803-2, Flow Diagram Auxiliary Feedwater, Rev. 32

2-47W803-3, Flow Diagram Auxiliary Feedwater, Rev. 25

Other

NPG-WBN2-3B-4002; Auxiliary Feedwater System Description; Rev. 2

FSAR Section 10.4.9; Auxiliary Feedwater System

Surveillance Instructions

2-SI-3-40; 18 Month Channel Calibration Steam Generator 1 Auxiliary Feedwater Flow Loop 2-LPF-3-163A; Rev. 1

2-SI-3-41; 18 Month Channel Calibration Steam Generator 1 Auxiliary Feedwater Flow Loop 2-LPF-3-155B; Rev. 1

2-SI-3-42; 18 Month Channel Calibration Steam Generator 1 Auxiliary Feedwater Flow Loop 2-LPF-3-147A; Rev. 1

2-SI-3-43; 18 Month Channel Calibration Steam Generator 1 Auxiliary Feedwater Flow Loop 2-LPF-3-170B; Rev. 1

2-SI-3-64; 18 Month Channel Calibration of Post Accident/Remote Shutdown Monitoring AFW Loop 2-LPF-3-147B; Rev. 1

2-SI-3-65; 18 Month Channel Calibration of Post Accident/Remote Shutdown Monitoring AFW Loop 2-LPF-3-155A; Rev. 1

2-SI-3-66; 18 Month Channel Calibration of Post Accident/Remote Shutdown Monitoring AFW Loop 2-LPF-3-163B; Rev. 1

2-SI-3-67; 18 Month Channel Calibration of Post Accident/Remote Shutdown Monitoring AFW Loop 2-LPF-3-170A; Rev. 1

2-SI-46-1; 18 Month Channel Calibration of Turbine Driven Auxiliary Feedwater Pump Flow Loops 2-LPF-3-142 And 2-LPF-46-57; Rev. 1

OA.1.10 Temporary Instruction 2515/152: Reactor Pressure Vessel Lower Head Penetration Nozzles and NRC Bulletin 2003-02: Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity

Reports

GE Inspection Technologies, Remote Visual Examination, Inspection Report, 09/03/2015

N-VT-17, Visual Examination for Leakage of PWR Reactor Head Penetrations, Form 1, Rev. 8, for Report No. R-P3748, 09/10/2015

Report No. R-P3748, Watts Bar Nuclear Plant Unit 2, Before Fuel Load, Reactor Pressure Vessel Lower Head and Bottom Mounted Instrument (BMI) Penetration Remote Visual VT-2 Examination, Final Report – September 2015, 09/10/2015

W47 150430 008, IEP-100, Administration of Nondestructive Examination (NDE) Procedures, Attachment 1, Temporary Change (TC), 04/30/2015

W47 130328 003, N-VT-17, Visual Examination for Leakage of PWR Reactor Head Penetrations, Rev. 8

Procedures

1-TRI-0-10.2, Watts Bar Nuclear Plant Unit 1, Technical Requirements Instruction, ASME Section XI ISI/NDE Program, Rev. 13

Watts Bar Nuclear Unit 2, Initial Preservice Exam (Before Fuel Load), RPV Lower Head VT-E – Resolution Check Verifications, 09/10/2015

Watts Bar Nuclear Unit 2, Initial Preservice Exam (Before Fuel Load), RPV Lower Head, Remote Visual (VT-E) Penetration Examination Results, Rev. 0

WBN-2 PSI, Preservice Inspection Program Plan, Rev. 9

Work Order 114906831, Perform bare metal visual inspection of the lower head penetrations on the WB2 RPV, 08/05/2015.

Certifications

Certificate No. IJ103000I, Certificate of Conformity for Visual Illumination Cards, Serial Nos. 301 through 800 (Serial No. 750 used), 09/11/2003

Procedure No. N-GP-9, Attachment 2, Visual Examination Card (VIC) Size Verification Form, Rev. 14 for PO/Release No. RLSE00055, Serial Nos. 301 through 800 (Serial No. 750 used), 09/28/2006

Training attendance records for Employee Identification No. 2YMXFPQOQ

W47 150824 003, Visual Acuity Examination Record for NDE/QC Personnel for Employee Identification No. 2YMXFPQOQ, 08/21/2015

W47 150811 004, Certificate of Method Qualification for Badge No. 2YMXFPQOQ, 07/16/2015

Drawings

ISI-2068A-E-12, Watts Bar Nuclear Plant Unit 2, Reactor Vessel Bottom Head Penetrations, Rev. 2

ISI-2068A-E-13, Watts Bar Nuclear Plant Unit 2, Reactor Vessel Bottom Head Penetrations, Rev. 1

OA.1.11 Construction Deficiency Report 391/84-34 Spurious Valve Operation Due to Fire Damage and OA.1.12 Construction Deficiency Report 391/84-35: Unacceptable Electrical Cable Separation

Work Orders and Completed Surveillance Records

WO 111182422, CCE DCN 54912 Stage 15 Rev. 1 SYS 067 070 213 WBN-2-MCC-B002-B

WO 111187960, CCE DCN 54912 Stage 14 Rev. 1 SYS 213 070 067 WBN-1-MCC-A002-A

WO 113817843, CGE DCN 55477 Stage 6 SYS 070 WBN-2-MVOP-070-156-A

WO 112210156, CCE DCN 54912 Stage 34 PIC 57881 SYS 070 213 278 2-FCV-070-0156-A

WO 111199573, CCE EDCR2 53217 PER 571361 PER 143750 SYS 001 WBN-2-HS-001-0147B-A

WO 112717280, CCE EDCR2 53217 PER 571361-004 SYS 001 WBN-2-HS-0148B-B

WO 110870586, CCE EDCR2 54870 SYS 063 2-HS-063-0072B-A

WO 112210156, CGE DCN 54912 Stage 34 PIC 57881 SYS 070 213 278 2-FCV-070-0156-A

Plant Modifications and Engineering Changes

EDCR 52945, Installs/Reroutes various cables

EDCR 53217, Install/Reroute various cables. Install local power disconnect switches for Steam Line Warming valves 2-FCV-1-147, 148, 149 and 150

EDCR 53287, Modify MCC bucket wiring to preclude spurious operation

EDCR 53421, Modify handswitch for valves 2-FCV-62-1228 and 1229. Modify supply circuit for valves 2-FCV-62-98 and 99 for in series breakers

EDCR 53580, Modify local valve control circuits to preclude spurious operation. Remove local Control stations for 2-FCV-63-25 and 26. Disconnect handswitch 2-HS-62-108B-A

EDCR 53756, Modify local valve control circuits for 2-FCV-68-332 and 333 to preclude spurious Operation

EDCR 54103, Adds internal jumpers to MOV's 2-FCV-68-332 and 333

EDCR 54144, Installs/Reroutes various cables. Add handswitch 2-HS-3-945 to remove power from 2-FCV-3-35, 35A, 48, 48A, 90, 90A, 103 and 103A

EDCR 54172, Add transfer switch with shorting contact for valves 2-FCV-30-17-A, -14-A, -56-A, -52-A, -59-A, -20-A, -40-A, -51-A, -7-A, -10-A and 2-FCV-30-8-B, -50-B, -9-B, -53-B, -15-B, -57-B, -16-B, -37-B, -19-B, -58-B to preclude spurious operation. Installs/Reroutes various cables. Install Junction Box in Turbine Bldg for storage of Spare parts for restoring RHR cooler fans and RHR valves.

EDCR 54255, Remove local control stations for 2-FCV-72-2, 39, 44 and 45 and revise local wiring to preclude spurious operation. Removed local control stations for Pump Motors 2-

MTR-72- 10 and 27

EDCR 54870, Remove local control stations for 2-FCV-63-72 and 73. Modify valve control circuits to preclude spurious operation

DCN 54912, Installs/Reroutes various cables. Delete local handswitches. Modify valve control circuits to prevent spurious operation. Stages 49, 50 & 62 Install SFP Pump Emergency Stop Handswitches

EDCR 54633, Installs conduits for App R cables

EDCR 54636, Installs/Reroutes various cables

EDCR 54637, Installs/Reroutes various cables

EDCR 54639, Installs/Reroutes various cables

EDCR 54640, Installs/Reroutes various cables

EDCR 55231, Installs/Replaces various cables

EDCR 55233, Installs/Replaces various cables

OA.1.13 EA-15-075 – Drilling Through Reinforcing Steel

Corrective Actions

PER# 635002, Problem Evaluation Report – Corrective Action Plan for Possible Improper Anchor Bolt Installation, 11/1/2012

CR# 838631, Potential Damaged Rebar, 3/31/2015

Miscellaneous

CNL-15-152, Watts Bar Nuclear Plant, Unit 2 – Reply to Notice of Violation (EA-15-075), dated: 7/24/2015

Commitment Completion Form, T02 150518 002, 10 CFR 50.9 Refresher Training, dated: 5/14/2015

Procedures

NC-PP-3, Watts Bar Unit 2 Corrective Action Program, Rev. 14

BPP-01.1, Administration of Site Procedures, Rev. 13

25402-000-GPP-0000-N1206, Work Order Processing, Rev. 20

NPG-SPP-03.1.6, Root Cause Analysis, Rev. 5

NPG-SPP-03.1.5, Apparent Cause Evaluations, Rev. 5

Work Orders

WO# 116723658 – CCI SYS 043 EDCR 53919 FCR 65288 AA 01 Stress Problem, dated: 5/27/2015

WO# 116771932 – CCI SYS 043 EDCR2 53917 FCR 65287 AA-04 WBN-2-HGR-043-AB Stress, dated: 6/18/2015

WO# 116771095 – CCM EDCR NONE SYS 077 WBN-2-HGR-077-RB, dated: 6/26/2015

WO# 116734756 – CCM EDCR2 54655 FCR 64995 AA-05 SYS 026 026.1 WBN-2-HGR-026-RB, dated: 5/15/2015

OA.1.14 Generic Letter 89-10: Safety-Related Motor Operated Valve Testing and Surveillance

Calculations

MDQ0029992015000687, Watts Bar Nuclear Plant Unit 2 JOG MOV Periodic Verification Classification, Rev. 1

MDQ00299920090310, Generic Letter 89-10 MOV Population for Watts Bar (Unit 2), Rev. 9

MDQ00200320080208, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-3-87, Rev. 3

MDQ00200320080211, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-3-126A, Rev. 4

MDQ00202620080219, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-26-243, Rev. 3

MDQ00206220080221, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-62-63, Rev. 3

MDQ00206220080225, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-LCV-62-133, Rev. 3

MDQ00206320080233, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-63-6, Rev. 7

MDQ00206320080234, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-63-7, Rev. 6

MDQ00206320080235, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-63-8, Rev. 7

MDQ00206320080236, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-63-11, Rev. 7

MDQ00206320080245, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-63-94, Rev. 8

MDQ00206320080249, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-63-157, Rev. 8

MDQ00207020080280, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-70-133, Rev. 5

MDQ00207020080281, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-70-134, Rev. 6

MDQ002077220080290, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-72-21, Rev. 4

MDQ00207220080294, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-72-40, Rev. 5

MDQ00207220080294, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-72-44, Rev. 3

MDQ00207420080301, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-74-8, Rev. 7

MDQ00207420080304, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-74-21, Rev. 3

MDQ00207420080306, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-74-33, Rev. 6

OA.1.15 Generic Letter 96-05: Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves

Procedures

0-MI-0.006, MOVATS Testing of Motor Operated Valves, Rev. 5

Work Orders

620700113, 2-MVOP-070-0089-B: Perform Static Diagnostic Testing (MOVATS) IAW 0-MI-0.006

620700114, 2-MVOP-070-0092-A: Perform Static Diagnostic Testing (MOVATS) IAW 0-MI-0.006

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620700118, 2-MVOP-070-0156-A, MOVATS Testing

620700119, 2-MVOP-070-0087-B: Perform Static Diagnostic Testing (MOVATS) IAW 0-MI-0.006

620700120, 2-MVOP-070-0133-A: Perform Static Diagnostic Testing (MOVATS) IAW 0-MI-0.006

620700121, 2-MVOP-070-0134-B: Perform Static Diagnostic Testing (MOVATS) IAW 0-MI-0.006

620700122, 2-MVOP-070-0183-A: Perform Static Diagnostic Testing (MOVATS) IAW 0-MI-0.006

620720065, 2-MVOP-072-0022-A, Perform Static Diagnostic Testing (MOVATS) IAW 0-MI-0.006
 620720066, 2-MVOP-072-0002-B, Perform Static Diagnostic Testing (MOVATS) per procedure 0-MI-0.006
 620720067, 2-MVOP-072-0013-B, Perform Static Diagnostic Testing (MOVATS) per procedure 0-MI-0.006
 620720068, 2-MVOP-072-0021-B, Perform Static Diagnostic Testing (MOVATS)
 620720069, 2-MVOP-072-0034-A, Perform Static Diagnostic Testing (MOVATS)
 620720070, 2-MVOP-072-0039-A, Perform Static Diagnostic Testing (MOVATS) per procedure 0-MI-0.006
 620720071, 2-MVOP-072-0040-A, Perform Static Diagnostic Testing (MOVATS) per procedure 0-MI-0.006
 620720072, Perform Static Diagnostic Testing (MOVATS) per procedure 0-MI-0.006 [for valve 2-MVOP-072-0041-B]
 620720073, Perform Static Diagnostic Testing (MOVATS) per procedure 0-MI-0.006 [for valve 2-MVOP-072-0044-A]
 620720074, Perform Static Diagnostic Testing (MOVATS) per procedure 0-MI-0.006 [for valve 2-MVOP-072-0045-B]

Miscellaneous

MDN-000-999-2011-0178, WBN Probabilistic Risk Assessment – Risk Ranking of Motor Operated Valves (MOV), Rev. 1
 T25 151006 031, Watts Bar Nuclear Plant, MOV Expert Panel (MOVEP) – Meeting Minutes, 12/10/2013 and 12/16/2013
 T25 151006 031, Attachment 4, Risk/Margin Matrix WBN Unit 1 – U1C11, 02/28/2013

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MDQ0029992015000687, Watts Bar Nuclear Plant JOG MOV Periodic Verification Classification, Rev. 1
 MDQ0020702008-0276, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-70-087, Rev. 1
 MDQ0020702008-0277, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-70-089, Rev. 3
 MDQ0020702008-0279, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-70-092, Rev. 3
 MDQ0020702008-0280, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-70-133, Rev. 5
 MDQ0020702008-0281, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-70-134, Rev. 6
 MDQ0020702008-0282, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-70-100, Rev. 3
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 MDQ0020702008-0288, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-72-002, Rev. 3
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 MDQ0020702008-0291, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-72-022, Rev. 3

MDQ0020702008-0292, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-72-034, Rev. 4
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MDQ0020702008-0295, Documentation of Design Basis Review, Required Thrust/Torque Calculations and Valve and Actuator Capability Assessment for Valve 2-FCV-72-41, Rev. 5
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LIST OF ACRONYMS

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Documents Access and Management System
AFW	Auxiliary Feedwater
ANI	Authorized Nuclear Inspector
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
CAPR	Corrective Actions to Prevent Recurrence
CAQR	Condition Adverse to Quality Report
CAR	Containment Air Return
CBI	Chicago Bridge & Iron Company
CCS	Component Cooling System
CDR	Construction Deficiency Report
CFR	Code of Federal Regulations
CILRT	Containment Integrated Leak Rate Test System
CLS	Central Laboratories Services
COMS	Cold Overpressure Mitigation System
CR	Condition Report
CRDM	Control Rod Drive Mechanism
CVCS	Chemical Volume Control System
DCN	Design Change Notice
DCS	Distributed Control System
DRA	Drawing Revision Authorization
ECN	Engineering Change Notice
EDCR	Engineering Document Construction Releases
EGTS	Emergency Gas Treatment System
EQCS	EQ Change Supplements
EQIR	Environmental Qualification Information Release
ERCW	Essential Raw Cooling Water
FCR	Field Change Requests
FOSAR	Foreign Object Search and Retrieval
FPR	Fire Protection Report
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GL	Generic Letter (NRC)
gpm	Gallons per minute
GWDS	Gaseous Waste Disposal System
HEPA	High-efficiency Particulate Air
HFT	Hot Functional Test
HMS	Hydrogen Mitigation System
IIR	Integrated Inspection Report
ILRT	Integrated Leak Rate Test
IMC	Inspection Manual Chapter (NRC)
IP	Inspection Procedure (NRC)
ISI	In-service Inspection
JOG	Joint Owner's Group
LHP	Lower Head Penetration
LLRT	Local Leak Rate Test
LOCA	Loss of Cooling Accident
LOOP	Loss of Offsite Power
LWR	Light Water Reactor
M&TE	Measuring and Test Equipment

MOV	Motor Operated Valve
NCR	Non-Conformance Report
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NIST	National Institute of Standards and Technology
No.	Number
NPP	Nuclear Performance Plan
NPSH	Net Positive Suction Head
NQAP	Nuclear Quality Assurance Plan
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OTO	One Time Only
PER	Problem Evaluation Report
PRA	Probabilistic Risk Assessment
PSI	Preservice Inspection
PSIG	Pounds per Square Inch Gauge
PTI	Preoperational Test Instruction
PWR	Pressurized Water Reactor
QA	Quality Assurance
QC	Quality Control
RBPVS	Reactor Building Purge Ventilating System
RCDT	Reactor Coolant Drain Tank
RCS	Reactor Coolant System
Rev.	Revision
RG	Regulatory Guide
RHR	Residual Heat Removal
RIS	Regulatory Issue Summary
RMS	Radiation Monitoring Systems
RPV	Reactor Pressure Vessel
S/G	Steam Generator
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SI	Safety Injection
SIT	Structural Integrity Test
SL	Severity Level
SME	Subject Matter Expert
SOI	System Operating Instruction
SP	Special Program
SRs	Surveillance Requirements
SSCs	Structures, Systems, and Components
SSD	Setpoint and Scaling Document
SSER	Supplemental Safety Evaluation Report
TDN	Test Deficiency Notice
TI	Temporary Instruction (NRC)
TMI	Three Mile Island
TRIs	Technical Requirements Instructions
TS	Technical Specifications
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
U2	Unit 2
WBN	Watts Bar Nuclear Plant
WB2	Watts Bar Nuclear Plant, Unit 2
WO	Work Order

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