Tennessee Valley Authority, Post Office Box 2000 Spring City, Tennessee 37381

April 22, 2016

10 CFR 50.59(d)(2)

U. S. Nuclear Regulatory Commission
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Watts Bar Nuclear Plant, Unit 1
Facility Operating License No. NPF-90
NRC Docket No. 50-390

Subject: **WATTS BAR NUCLEAR PLANT UNIT 1 – TITLE 10, CODE OF FEDERAL REGULATIONS 50.59 SUMMARY REPORT**

Reference: Letter from TVA to NRC, "Watts Bar Nuclear Plant Unit 1 - Title 10, Code of Federal Regulations 50.59 Summary Report," dated November 3, 2014

Pursuant to Title 10, *Code of Federal Regulations* (10 CFR) 50.59(d)(2), the Tennessee Valley Authority (TVA) is submitting a summary report of the changes, tests, and experiments implemented at Watts Bar Nuclear Plant (WBN) Unit 1 since the last 10 CFR 50.59 report was submitted to the Nuclear Regulatory Commission (NRC) in the referenced letter. The evaluations summarized in the enclosure, which cover the period from May 2, 2014 through October 22, 2015, demonstrate that the described changes do not meet the criteria for license amendments as defined by 10 CFR 50.59(c)(2).

There are no new regulatory commitments in this letter. Should you have questions regarding this submittal, please contact Gordon Arent, Director of Watts Bar Site Licensing, at (423) 365-2004.

Respectfully,

Paul Simmons
Site Vice President
Watts Bar Nuclear Plant

Enclosure:

Watts Bar Nuclear Plant Unit 1 10 CFR 50.59 Summary Report

cc (Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant, Unit 1
NRR Project Manager - Watts Bar Nuclear Plant

ENCLOSURE
WATTS BAR NUCLEAR PLANT UNIT 1
10 CFR 50.59 SUMMARY REPORT

1. Evaluation: Design Change Notice (DCN) 57326A, R0
2. Evaluation: DCN 63490A, R0
3. Evaluation: DCN 63651A, R0
4. Evaluation: Interface Boundary Clearance Release (IBCR) -14-065-001, R1
5. Evaluation: IBCR-14-067-3, R2
6. Evaluation: Permanent Boundary Change (PBC) -13-067-002, R1
7. Evaluation: Safety Analysis Report (SAR) Change Package 13-008, R0
8. Evaluation: Procedure Change FHI-12 Rev. 9, Evaluation R0
9. Evaluation: Temporary Modification (TMOD)-WBN-0-2014-030-002 Rev. 2, Evaluation R0
10. Evaluation: TMOD-WBN-0-2014-030-003 Rev. 1, Evaluation R1
11. Evaluation: TMOD-WBN-1-2014-043-003 Rev. 0, Evaluation R1
12. Evaluation: TMOD-WBN-0-2014-044-004 Rev. 2, Evaluation R0
13. Evaluation: TMOD-WBN-0-2015-030-001, Evaluation R0
14. Evaluation: WBN Unit 1 Cycle 14 Core Operating Limits Report (COLR) R0

1. EVALUATION: DESIGN CHANGE NOTICE (DCN) 57326A, R0

DCN 57326 Replaces Station Control Air System (CAS) air compressors 0-COMP-32-0025 (Compressor A), -0026 (Compressor B), and -0027 (Compressor C) and air dryers 0-DRYR-032-0010 and -0011 (Dryer A), 0- DRYR-032-0015 and -0016 (Dryer B), 0-DRYR-032-0156 and -0157 (Dryer C), and associated components. This DCN also replaces existing analog compressor sequencing control with local digital controls for each compressor. The reason for the design change is primarily to improve plant reliability through replacement of older equipment with new equipment of equivalent or better design.

Based on 10 CFR 50.59 Screening Review, Question 1 was found to adversely affect an Updated Final Safety Analysis Report (UFSAR) described design function. UFSAR Section 9.3.1.5 states the control air system is designed to operate automatically. The automatic operation of the new compressors will be maintained with the new Intellisys controllers, which are considered digital equipment; however, sequencing of the compressors has to occur manually at each compressor. The existing sequencing panel automatically changes the set pressure for each compressor, while the new Intellisys controllers are required to be manually set at each compressor. Converting a feature that was automatic to manual is potentially adverse. Additionally, the change from the sequencing panel is potentially adverse as the change creates new potential failure modes in the interaction of operators with the system as there are more steps to change the sequencing of the compressors due to the proposed change.

Summary of Evaluation:

New Station Control Air Compressors and dryers are replacing the existing compressors and dryers A, B and C. Each new compressor and dryer operates independently from the others. Failure of one compressor or dryer will not impact operation of the remaining compressors or dryers. Each of the compressors is controlled by identical digital control devices supplied by the vendor, Ingersoll Rand. If the software/firmware version loaded onto these digital controllers contains a software flaw or logic fault that would result in compressor shutdown, then whatever triggers this fault could potentially result in a loss of all three of the new compressors. In addition, each dryer is controlled by identical digital control devices supplied by the vendor, Pneumatic Products/SPX. If the software/firmware version loaded onto the digital controllers contains a software flaw or logic fault that would result in dryer shutdown, then the fault could potentially result in a loss of all three dryers. It should be noted that Compressors A, B and C normally function as a backup and /or supplement to the Compressor D which, under normal plant conditions, provides all of the control air needs for the plant. This potential common software/firmware fault would still only disable the A, B or C compressors or A, B and C dryers, not the entire CAS system. The operating ranges of the equipment are within the Turbine Building area temperatures and humidity.

The new compressors will be manually re-sequenced per plant procedure by changing Compressor A, B and C operating set points. The periodic re-sequencing does involve a moderate human performance issue but is expected to only potentially affect a single compressor setpoint at a time. Therefore, the proposed change does not adversely affect a UFSAR described design function due to the digital upgrade.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

2. EVALUATION: DCN 63490A, R0

The Condenser Vacuum Exhaust (CVE) Radiation Monitors continuously monitor the mechanical vacuum pump air exhaust for an indication of a primary-to-secondary leak and to provide indication in the event of a steam generator tube rupture (SGTR). The proposed activity is a DCN modification to replace the obsolete CVE post accident effluent radiation monitors in the Unit 1 Turbine Building and associated equipment in the Main Control Room. The capabilities of the new monitors will be equal to or better than the existing monitors in Unit 1 while maintaining similarity to radiation monitors in other TVA Nuclear plants.

The new configuration will consist of a different style of detectors that have lower maintenance. Two (2) different types of detectors shall be installed to perform the same function of 1-RE-90-404. This new configuration will consist of two new area monitor detectors (1-RE-90-255 and 1-RE-90-256) that will be mounted directly facing the Condenser Vacuum Pump Exhaust pipe. The two (2) detectors, 1-RE-90-255 and 1-RE-90-256 will replace the function of the two channels (404A and 404B) of 1- RE-90-404. 1-RE-90-255 will be a Geiger-Muller tube which will monitor the mid-range condenser vacuum exhaust while 1-RE-90-256 will be a gas-filled gamma ionization chamber which will monitor the condenser vacuum exhaust high range. The loops associated with 255 and 256 will have better reliability compared to the existing 404 loop as the instrument type is not affected by moisture in the lines and is less susceptible to temperature related accelerated aging. Two new RM-1000 digital radiation processors will provide local indication. The RM-1000s will also process the signals from the detectors to provide indication and alarms to the operators through the Integrated Computer System (ICS) and annunciators.

Summary of Evaluation:

DCN 63490 implements a digital modification consistent with current industry guidance on licensing digital upgrades. This DCN encompasses both hardware and software changes to the CVE Radiation Monitoring System.

The proposed changes identified under this DCN do not adversely impact how the UFSAR described design function is performed.

Equipment additions addressed in this DCN have been evaluated with regards to suitability of the installed location, along with any associated environmental impact (e.g., temperature, humidity, seismic, electromagnetic fields, Electro Magnetic Interference/Radio Frequency Interference [EMI/RFI] emissions, 'heat load' margin). No adverse interactions were identified, which could lead to equipment performance degradation in either: (1) equipment to be installed or replaced per this DCN or (2) an existing plant SSC in the surrounding area.

Each evaluation contained in this report supports the conclusion that DCN 63490 implementation has a negligible effect on: (1) the frequency of occurrence of previously analyzed Chapter 15 accidents; (2) the likelihood of occurrence of a malfunction of an SSC important to safety, previously evaluated in the UFSAR; (3) the consequences of an accident previously evaluated in the UFSAR; (4) the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR; (5) the possibility for an accident of a different type than any previously evaluated in the UFSAR; (6) the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR; (7) a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered; or (8) a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

3. EVALUATION NUMBER: DCN 63651A, R0

Watts Bar Nuclear Plant is to design and utilize Foreign Material Exclusion (FME) covers to keep foreign material out of Tritium Production Burnable Absorbing Rod (TPBAR) canisters stored in the Spent Fuel Pool.

Summary of Evaluation:

FME covers were evaluated for effects on the Spent Fuel Racks, the TPBARs, and long term (more than 18 month) storage of TPBARs. Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

4. EVALUATION: Interface Boundary Clearance Release (IBCR)-14-065-001, R1

This IBCR for the Emergency Gas Treatment System (EGTS) discusses opening Unit 1/Unit 2 interface valves and performance of Unit 2 EGTS testing. Performance of Unit 2 EGTS Air Cleanup subsystem (ACU) testing creates a potential for the Unit 1 post-Loss of Coolant Accident (LOCA) EGTS ACU function to be compromised.

Summary of Evaluation:

The evaluation considers the impact caused by testing Unit 2 EGTS ACU function while Unit 1 EGTS ACU is required to remain operable to function post LOCA. Testing Unit 2 EGTS ACU while Unit 1 EGTS ACU remains Technical Specification operable has been analyzed, but the calculation contains a Unit 2 annulus inleakage assumption equivalent to the future Unit 2 Technical Specification requirement. Since inleakage won't be verified at the start of initial Unit 2 EGTS ACU testing, the calculation cannot be used. Therefore, manual operator actions are required to isolate Unit 2 EGTS ACU following receipt of a Unit 1 Phase A Isolation prior to inleakage verification. Once Unit 2 annulus inleakage and current Unit 1 EGTS ACU Technical Specification requirements are verified to be within specifications, the calculation shows that Unit 2 EGTS ACU testing may occur at the same time Unit 1 EGTS function is maintained without the need for manual operator actions. These actions are considered temporary for the initial testing of Unit 2 EGTS.

The Unit 2 EGTS ACU dampers that will be required closed following receipt of a Unit 1 Phase A Isolation Signal are shown below. Prior to performance of the test and reliance on these valves to close, they will be tested to the point that they are shown functional from the control room. The temporary use of manual operation meets the requirements of a temporary manual action under the guidance provided in NEI 96-07, Revision 1.

2-FCO-065-0045-B	2-FCV-065-0029-B	2-FCV-065-00207-B
2-FCO-065-0046-B	2-FCV-065-0050-A	2-FCV-065-0009-A
2-PCV-065-0081-A	2-PCV-065-0086-A	
2-PCV-065-0083-B	2-PCV-065-0087-A	

There is no increase in the frequency of occurrence of accident or increase in the likelihood of malfunctions evaluated in the UFSAR. The activity does not result in any offsite or main control room dose changes, new release paths, changes to the fuel cladding, Reactor Coolant System

changes, or changes to primary containment. This change is not the initiator of any new accident nor does it result in a malfunction with a different result. There are no new failure modes identified from this test and there is not a change in evaluation methodology.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

5. EVALUATION: IBCR-14-067-3, R2

This IBCR for the Essential Raw Cooling Water (ERCW) System opens the Unit 1/Unit 2 interface valves. The opening of these valves will allow water flow into the Unit 2 areas. Allowing the flow from the common ERCW system into the Unit 2 areas will potentially cause a reduction of the flow rates to the Unit 1 equipment.

Summary Evaluation:

The evaluation considers the impact caused by requiring the normally isolated trained Unit 1 and Unit 2 Containment Spray HX ERCW supply and return isolation valves being opened to ERCW flow while the associated train is being tested. Opening ERCW flow to a Containment Spray HX during testing or temporary operation has been analyzed, but this has not been analyzed during other non-normal alignments. Therefore, this test and temporary operations require that operators be stationed to isolate the opened Unit 1 or Unit 2 Containment Spray HX ERCW supply and return isolation valves at the initiation of any non-normal event or accident.

The Containment Spray HX valves that will be opened with a dedicated main control room (MCR) operator on Train B are 1-FCV-067-0123-B and 1-FCV-067-0124-B, 2-FCV-067-0123-B and 2-FCV-067-0124-B. The valves that will be opened with a dedicated operator on Train A are 1-FCV-067-0125-A and 1-FCV-067-0126-A or 2-FCV-067-0125-A and 2-FCV-067-0126-A. These valves are operable from the control room.

When flow is established to both Lower Containment Vent Coolers 2A and 2C, Supply isolation valve 2-ISV-67-523A may be left unattended provided that the downstream "A" Train throttle valves are in the as left test position and, prior to completion of the dual Unit ERCW flow balance, the "A" Train ERCW pumps are able to provide at least 98% of their tested acceptance flows and ERCW flow through the CCS HX A and B is no greater than normal Unit 1 only operational flows. If these conditions are not met or in place, the isolation valve shall either be closed or be manned by a dedicated operator.

When flow is established to both Lower Containment Vent coolers 2B and 2D, Supply isolation valve 2-ISV-67-5238 may be left unattended provided that the downstream "B" Train throttle valves are in the as left test position. If this condition is not in place, the isolation valve shall either be closed or be manned by a dedicated operator.

After completion of the dual Unit ERCW flow balance, the IBCR branch lines (with exception to the Unit 2 Containment Spray Heat Exchangers) may be unisolated and unattended as long as the Unit 2 throttle valves are in their as left flow balance test position.

There is no increase in the frequency of occurrence of accident or increase in the likelihood of malfunctions evaluated in the UFSAR. The activity does not result in any offsite or main control room dose changes, new release paths, changes to the fuel cladding, Reactor Coolant System changes, or changes to primary containment. This change is not the initiator of any new

accident nor does it result in a malfunction with a different result. There are no new failure modes identified from this test and there is not a change in evaluation methodology.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

6. EVALUATION: Permanent Boundary Change (PBC) -13-067-002, R1

This PBC for the ERCW System opens the Unit 1/Unit 2 interface valves. The opening of these valves will allow water flow into the Unit 2 areas. Allowing the flow from the common ERCW system into the Unit 2 areas will potentially cause a reduction of the flow rates to the Unit 1 equipment.

Summary of Evaluation:

The evaluation considers the impact caused by requiring the normally isolated trained Unit 1 and Unit 2 Containment Spray Heat Exchangers ERCW supply and return isolation valves being opened to ERCW flow while the associated train is being tested. Opening ERCW flow to a Containment Spray Heat Exchanger during testing has been analyzed, but this has not been analyzed during other non-normal alignments. Therefore, this test requires that operators be stationed to isolate the opened Unit 1 or Unit 2 Containment Spray HX ERCW supply and return isolation valves at the initiation of any non-normal event or accident.

The Containment Spray HX valves that will be opened with a dedicated MCR operator to close during Train B testing are 1-FCV-067-0123-B and 1-FCV-067-0124-B, 2-FCV-067-0123-B and 2-FCV-0670124-B. The valves that will be opened with a dedicated operator to close during Train A testing are 1-FCV067-0125-A and 1-FCV-067-0126-A or 2-FCV-067-0125-A and 2-FCV-067-0126-A. These valves are operable from the control room.

When flow is established to both Lower Containment Vent Coolers 2A and 2C, Supply isolation valve 2-ISV67-523A may be left unattended provided that the downstream "A" Train throttle valves are in the as left test position, the "A" Train ERCW pumps are able to provide at least 98% of their tested acceptance flows and ERCW flow through the CCS HX A and B is no greater than normal Unit 1 only operational flows. If these conditions are not met or in place, the isolation valve shall either be closed or be manned by a dedicated operator.

When flow is established to both Lower Containment Vent coolers 2B and 2D, Supply isolation valve 2-ISV-67523B may be left unattended provided that the downstream "B" Train throttle valves are in the as left test position. If this condition is not in place, the isolation valve shall either be closed or be manned by a dedicated operator.

There is no increase in the frequency of occurrence of accident or increase in the likelihood of malfunctions evaluated in the UFSAR. The activity does not result in any offsite or main control room dose changes, new release paths, changes to the fuel cladding, Reactor Coolant System changes, or changes to primary containment. This change is not the initiator of any new accident nor does it result in a malfunction with a different result. There are no new failure modes identified from this test and there is not a change in evaluation methodology.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

7. EVALUATION: SAR Change Package 13-008, R0

The proposed activity substitutes the current Westinghouse nuclear fuel assembly cladding corrosion model with a new cladding corrosion model, which was approved by the NRC in July 2013, as documented in WCAP-12610-P-A Addendum 2-A. The Integral Form ZIRLO Cladding Corrosion Model improves the concept of modified fuel duty index (MFDI) and incorporates the latest fuel cladding corrosion measured oxide thickness data. Westinghouse has forward fit all new fuel rod analyses with the new model since the approval of the topical report including the analysis conducted for Watts Bar Unit 1, starting with Unit 1 Cycle 14 reload design.

Summary of Evaluation:

A 50.59 evaluation is required because the method of evaluation described in WCAP-12610-P-A Addendum 2-A is replacing the current method described in the UFSAR. It is shown that the change does not constitute departure from a method of evaluation and, therefore, that a license amendment is not required prior to implementation of the change. Use of the Integral Form ZIRLO Cladding Corrosion Model does not constitute a departure because (1) the Integral Form ZIRLO Cladding Corrosion Model is approved by the NRC specifically for PWR fuel rod design analyses and (2) the Integral Form ZIRLO Cladding Corrosion Model was used under the terms, conditions, and limitations of that NRC approval.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

8. EVALUATION: Procedure Change FHI-12 Rev. 9, Evaluation R0

Revision 9 to FHI-12, "Primary source Installation," will change the method by which primary source rodlets are brought on site due to being shipped in a Frontier 50300 Type A cask as opposed to the cask used in Revision 8 to bring in the first Unit 1 primary source rodlets. This procedure will also block Spent Fuel Pool Accident Radiation Monitors to prevent Auxiliary Building Isolation actuation. Other evaluated changes include improved tooling from the vendor since the procedure was last performed.

Summary of Evaluation:

Blocking the Spent Fuel Pool Accident Radiation Monitors during source rodlet movement and installation does not result in an unreviewed safety question. The accident mode evaluated in the UFSAR is for a postulated fuel handling accident. Specifically, the accident case where a fuel handling accident occurs in the spent fuel pit/Auxiliary Building with no Auxiliary Building Isolation (ABI) and with unfiltered releases through the Auxiliary Building vent. (UFSAR Section 15.5.6) The proposed procedure revision to allow source rodlet handling activities with Spent Fuel Pool Accident Radiation Monitors blocked to prevent Auxiliary Building Isolation is bounded by the fuel handling accident because the radioactive material within the source rodlet is not an airborne substance which does not pose the same concerns as a fuel handling accident involving irradiated fuel. No irradiated fuel will be handled with the radiation monitors' ABI signal blocked, therefore the probability of a fuel handling accident in this configuration cannot increase.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

9. EVALUATION: Temporary Modification (TMOD)-WBN-0-2014-030-002 Rev. 2, Evaluation R0

Unit 2 desires to operate the Unit 2 Containment Purge system (U2 CPS). Certain U2 CPS valves perform a Unit 1 interim Auxiliary Building Secondary Containment Enclosure (ABSCE) isolation function and their permanent Unit 1 design requires control air to them to be isolated and the valves failed closed. Those valves and other U2 CPS valves and components have been temporarily modified in accordance with TMODS 0-2014-030-002 R1 and 0-2014-030-003 R1 to provide redundant automatic ABSCE isolation upon receipt of ABI or High Rad in Refueling Area (HRRA) isolation signals. Revision 2 of this TMOD changes the configuration of the control circuits of the valves and fans that receive the isolation signal. The revision 2 configuration includes the Unit 2 Solid State Protection System (SSPS) slave relay contacts in series with the temporary relay contacts installed by revision 1 of this TMOD. The control circuits of the coils of the Unit 2 SSPS relays are not modified and are under Unit 2 control. This review only addresses the Unit 1 ABSCE function. The scope of the review is adding the control circuits to Unit 1 and modifying them so that Unit 1 does not rely on Unit 2 SSPS to isolate ABSCE. There are no ductwork modifications in the scope of this TMOD, they are in TMOD 0-2014-030-003.

The fans that stop in response to the control signal are not safety related and are not required to stop to protect the ABSCE boundary. The population of valves is the system 30 valves that will in the future receive a Unit 2 Train A Containment Ventilation Isolation (CVI) signal. At present the Unit 2 reactor building is open to the environment. A TMOD is implementing the configuration because the Unit 2 SSPS, which is intended to be part of the automation scheme in the future, is not available for service.

Unit 1 DCN 52220-A modified the Train A and Train B electrical circuits of the high radiation in the refueling area logic bus and the Train A and Train B SSPS input to create the so called "ABI/CVI" design. Unit 2 has the same design. TMOD 0-2014-030-002 modifies the Train A electrical circuits for ABI and HRRA. The modification is implemented in Auxiliary Relay Rack 2-R-73 which is operating in support of Unit 1 and Unit 2 SSPS rack 2-R-48 which is not in active Unit 2 service but is qualified to Seismic category 1 standards and owned by Unit 1.

Components that have been designed, procured, and installed by Unit 2 to equivalent standards as Unit 1 will be tested and placed in service for Unit 1. Any outstanding Unit 2 construction work will be evaluated for its impact on the proper operation of the equipment. The actuating signal originates in Auxiliary Relay Rack 1-R-73 and is sent to 2-R-73 on a cable originally intended to carry the Unit 2 CVI signal. In rack 2-R-73 the signal passes through 2-HS-90-410, which will be placed under unit 1 control. When 2-HS-90-410 is in the refuel position the signal will actuate two temporary Cutler Hammer type AR relays that are equivalent to Unit 2 SSPS slave relays K615 and K622. The relays' contacts are wired to temporary cables in temporary conduits routed to Unit 2 SSPS rack 2-R-48. The cables and conduit are designed to permanent design standards and electrical loading and voltage drop calculations have been performed.

Within 2-R-48 the temporary relay contacts are wired in series [normally closed (NC) contacts open upon actuation signal) with the permanent Unit 2 slave relays. If Unit 2 activities actuate the permanent relays the components controlled by them will go to the accident position but will not affect Unit 1 operation. 2-HS-90-410 must be in the Refuel position when the automatic isolation function is required. The design of the train B controls are similar to train A and is described in detail in TMOD 0-2014-030-003.

The Unit 2 isolation valves, including the ductwork and dampers, were designed and qualified to perform the same functions as their Unit 1 counterparts and the same or equivalent design, fabrication, and construction standards were used. It is noted that the isolation valves are ASME Section III Class 2 or 3 and qualified to Seismic Cat 1 requirements, the dampers are mounted in the ductwork to Seismic Cat 1 requirements, and the ductwork/ supports has been addressed and qualified under the Unit 2 HVAC Duct and Duct Supports CAP.

A Technical Evaluation (TE) was performed for this modification and is included in the TMOD. The TE determined this activity complies with the safety and functional requirements specified in the applicable design basis documents and does not adversely affect the performance of any safety related equipment. This change does not introduce any new failure modes. Therefore, based on compliance with established design basis requirements, this change is safe and acceptable from a nuclear safety standpoint.

Credible failures associated with implementation of TMOD 0-2014-030-002 include failure of the affected relays and associated handswitches. The primary relevant design basis accidents are those that could cause an ABI.

Summary of Evaluation:

The proposed temporary modifications do not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction, or create a new type of accident. The design basis fission product barriers will not be altered or exceeded. No new method of evaluation was used in evaluating the proposed temporary modification.

As a result of this valuation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore, obtaining prior NRC approval is not required to implement this activity.

10. EVALUATION: TMOD-WBN-0-2014-030-003 Rev. 1, Evaluation R1

Unit 2 desires to operate the Unit 2 CPS (U2 CPS). Certain U2 CPS isolation valves presently have control air removed and are on the interim ABSCE boundary. Operation of those valves is automated by this TMOD. They are all train A valves and will respond to ABI and HRRA signals when 2-HS-90-410 is in the Refuel position. Additional U2 CPS valves, some train A, others train B will have that same automatic isolation function in the TMOD 0-2014-030-003 configuration and in the final ABSCE configuration. The train B valves are interlocked with 2-HS-90-415. TMOD 0-2014-030-002 modified the control circuits of the U2 CPS train A isolation valves but did not authorize opening the valves. This evaluation includes authorization to open all of the valves that are presently designed to receive a Unit 2 CVI signal. This review only addresses the unit 1 ABSCE function. The scope of the review is automation of the ABI/HRRA isolation function of the valves required to provide redundant ABSCE isolation in the interim and final ABSCE configurations and ensuring the pressure boundary between and including the valves is qualified. This TMOD includes all mechanical components and civil features required to qualify the pressure boundary. Specifically, the Secondary and Inboard (located in the Annulus) Containment Purge Isolation valves, all the ductwork between these valves, and the manual insert mounted balancing dampers located in the ductwork. This TMOD and TMOD 0-2014-030-002 also include valves and ductwork not credited as ABSCE components. They are included to ensure Unit 2 activities will not adversely affect the ABSCE components. TMOD 0-2014-030-002 is a prerequisite to this TMOD. Upon completion of the configuration changes authorized by this TMOD and post-installation testing, the U2 CPS ABSCE isolation features will be fully functional. Two TMODs are used so that only one train of actuation logic is inoperable

at a time. A TMOD is implementing the configuration because the unit 2 SSPS, which is intended to be part of the isolation scheme, is not available for service.

Summary of Evaluation:

Unit 1 DCN 52220-A modified the Train A and Train B electrical circuits of the high radiation in the refueling area logic bus and the Train A and Train B SSPS input to create the so called "ABI/CVI" design. Unit 2 has the same design. TMOD 0-2014-030-003 modifies the Train B electrical circuits for ABI and HRRA. The modification is implemented in Auxiliary Relay Rack 2-R-78 which is operating in support of Unit 1 and Unit 2 SSPS rack 2-R-51 which is not in active Unit 2 service but is qualified to Seismic category 1 standards and owned by Unit 1. Components that have been designed, procured, and installed by Unit 2 to equivalent standards as Unit 1 will be tested and placed in service for Unit 1. Any outstanding Unit 2 construction work will be evaluated for its impact on the proper operation of the equipment. The actuating signal originates in Auxiliary Relay Rack 1-R-78 and is sent to 2-R-78 on a cable originally intended to carry the Unit 2 CVI signal. In rack 2-R-78 the signal passes through 2-HS-90-415, which will be placed under Unit 1 control. When 2-HS-90-415 is in the refuel position the signal will actuate two temporary Cutler Hammer type AR relays that are equivalent to Unit 2 SSPS slave relays K615 and K622. The relays' contacts are wired to temporary cables in temporary conduits routed to Unit 2 SSPS rack 2-R-51. The cables and conduit are designed to permanent design standards and Electrical loading and voltage drop calculations have been performed. Within 2-R-51 the temporary relay contacts are wired in series (NC contacts open upon actuation signal) with the permanent Unit 2 slave relays. If Unit 2 activities actuate the permanent relays there will be no affect on the temporary circuits and there will be no affect on Unit 1 operation. 2-HS-90-415 must be in the Refuel position when the automatic isolation function is required. The design of the train A controls are similar to train B and is described in detail in TMOD 0-2014-030-002.

The Unit 2 isolation valves, including the ductwork and dampers, were designed and qualified to perform the same functions as their Unit 1 counterparts and the same or equivalent design, fabrication, and construction standards were used. It is noted that the isolation valves are ASME Section III Class 2 or 3 and qualified to Seismic Cat 1 requirements, the dampers are mounted in the ductwork to Seismic Cat 1 requirements, and the ductwork/supports has been addressed and qualified under the Unit 2 HVAC Duct and Duct Supports CAP.

A TE was performed for this modification and is included in the TMOD. The TE determined this activity complies with the safety and functional requirements specified in the applicable design basis documents and does not adversely affect the performance of any safety related equipment. This change does not introduce any new failure modes. Therefore, based on compliance with established design basis requirements, this change is safe and acceptable from a nuclear safety standpoint. Credible failures associated with implementation of TMOD 0-2014-030-003 include failure of the affected relays and associated handswitches. The primary relevant design basis accidents are those that could cause an ABI.

Summary of Evaluation:

The proposed temporary modifications do not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction, or create a new type of accident. The design basis fission product barriers will not be altered or exceeded. No new method of evaluation was used in evaluating the proposed temporary modification.

As a result of this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore, obtaining prior NRC approval is not required to implement this activity.

11. EVALUATION: TMOD-WBN-1-2014-043-003 Rev. 0, Evaluation R1

This TMOD is being performed in response to Problem Evaluation Report (PER) 881838 which identified a leaking compression fitting located just outside the crane wall in Accumulator Room #4. The fitting is located on the Pressurizer Gas Sampling Line, which is part of the routine sampling subsystem of the Sampling and Water Quality System (SWQS).

Previous attempts to stop the leak under WO# 115778035 were unsuccessful. The compression fitting is located upstream up 1-FCV-43-2 (Inside Containment Isolation Valve) and 1-SMV-43-673. Upstream of the fitting is a normally closed isolation valve (1-FCV-43-1) and a sample line root valve (1-SMV-68-576). Because the closed isolation valve is leaking by and the root valve cannot be manually manipulated while online, another alternative must be pursued.

The scope of this change is limited to sealing the singular compression fitting upstream of 1-SMV-43-673 and tagging out 1-FCV-43-1 and 1-FCV-43-2 in the closed position as detailed in TMOD WBN-1-2014-043-003. It will be sealed using a box-type enclosure with a sealant compound (X36 and G Fiber) injected. Implementation of the proposed activity will seal the fitting until it can be replaced during the next RFO. This activity will meet the requirements of Temporary Leak Repair Program (0-MI-0.031). Per section 3.1 (B) of this MI, the activity must meet the requirements of General Engineering Specification: G-29B, PS 4.M.4.3.

The X-36 sealant is an elastomer that can withstand the range from atmospheric to Pressurizer design pressure (2485 psig). The drawback is that the sealant is only rated to 450°F (G-Fiber rated to 750°F). Based on the vendor's input, this is the highest temperature rated elastomer that can handle fluctuations in pressure. Since the design temperature for the Pressurizer is and 680°F, this T-Mod will ensure the enclosure and sealant will not see temperatures greater or equal to 450°F. This will be accomplished by verifying the temperature is < 450°F prior to installation of the sealant and tagging out in the closed position 1-FCV-43-1 and 1-FCV-43-2 for the remainder of the cycle, until RCS temperatures are below 450°F. Chemistry was consulted and stated that this line is not currently used for sampling. Furthermore, degassing activities can still take place through the normal letdown process until the RCS temperature is below 450°F at which point the sample line can be utilized.

The Pressurizer Gas Sample Line acts as an air cooled heat exchanger, resulting in natural convective cooling as you move further from the Pressurizer. The sealant is justified for use below the design temperature of RCS based on the following. With valves 1-FCV-43-1 and 1-FCV-43-2 tagged in the closed position, and 1-FCV-43-1 leaking by minimally, a dead leg will exist up to 1-FCV-43-2 causing the temperature at the compression fitting to stabilize. The temperature will stabilize to ambient conditions. For the location of the compression fitting in Accumulator Room #4, the abnormal maximum temperature is 130°F. This is well below the rated temperature of the sealant. Additionally, a Post Maintenance Test (PMT) will be performed to verify the temperature at the compression fitting is below 450°F with the valve line up discussed above. Additionally, walkdown verified the temperature of the line was at approximately 108°F under the conditions in which the enclosure and sealant will be installed.

The X-36 sealant has no explicit pressure rating listed in the vendor materials. The X-36 sealant will be mixed in a blend with G-Fiber to increase its bridging characteristics. Per

correspondence from TEAM Industrial Industries, this sealant blend is suitable for this application. The memo and supporting TEAM data sheets are included as Attachment 3 of the TMOD.

This activity is treated as a housekeeping measure. For this activity, the sealant is used as a gasket material to stop the leak and not replace the pressure boundary. This is not a code repair.

Summary of Evaluation:

Implementation of this activity will temporarily affect the UFSAR described design function for the Pressurizer Gas Sample Line only, by taking it out of service. However, this sample location is utilized by the plant only on an infrequent basis during normal operations as a convenience to aid in locating potential reactor coolant system leaks. Chemistry was consulted and use of this line is not required during operation. Pressurizer liquid space samples can still be obtained through standard methods (1-CM-6.21).

The Pressurizer gas sample point is listed in the UFSAR (Table 9.3-2), and although there is no regular requirement to utilize this sample point, this activity will prohibit its use should a need arise during the operating cycle. Additionally, degassing activities will have to be performed during normal letdown versus utilizing the sample line until RCS temperature is below 450°F. Chemistry was contacted and existing procedures allow for alternate methods of degassing though the normal letdown process. Implementation of this TMOD does not involve a change to a procedure but does take a UFSAR described sample location out of service. Administratively tagging isolation valves 1-FCV-43-1 and 1FCV-43-2 in the closed position will prevent Pressurizer Gas Space samples from being taken at power. This is not a safety-related or required function, and containment integrity will be maintained. The only UFSAR described safety function for the Pressurizer Gas Sample Line is containment isolation. This function will not be impacted by this temporary modification. This activity will not affect any design basis accident response or credible failure modes.

Therefore, based on the above, this activity may be done with no adverse impact to nuclear safety nor the health and safety of the public. If the box-type enclosure with sealant were to fail, containment isolation would still be achieved.

Directly Affected SSCs:

The Pressurizer Gas Sample Line and all components downstream of 1-FCV-43-1 are directly affected. Isolation valve 1-FCV-43-1 and 1FCV-43-2 will be tagged out in the closed position for the remainder of the cycle until RCS temperatures are below 450°F. The compression fitting will be replaced off-line during the next RFO. Consequently, the Pressurizer Gas Sample Line and corresponding sample sink will not be operable until the fitting is replaced off-line. Additionally, the Pressurizer Gas Sample Line cannot be used for degassing until RCS temperatures are below 450°F.

Indirectly Affected SSCs:

Sealing the leaking compression fitting will positively impact the integrity of the Pressurizer Gas Sample Line. It will prevent any RCS fluid from leaching out in the Accumulator Room until the fitting can be replaced. This sample line is not required for any design basis accidents and this activity will not adversely impact any credited failure modes. There are no failure modes discussed or identified in Section 9.3.2 of the FSAR. The failure mode of the sealant and

box enclosure, per the vendor, is as follows. *When X-36 Bonder is exposed to excessive temperature (greater than 450°F), evidence of degradation will be the release of nuisance smoke and fumes. This will result in a porous solid without any mechanical integrity or cohesiveness. It would be expected that the reduction in strength will allow the service to escape and a leak will occur. The fumes are not expected to be toxic but can be irritating causing respiratory distress.* The sealants that are being used are X-36 and G-Fiber. X-36 is a Silicone Rubber while G-fiber is an Aluminum Silicate. The calculated amount of sealant to be added is only 0.05 lbs. This calculation was done by TEAM Industrial Services and is included in Attachment 1 of this T-MOD. This amount is less than the equivalent hazard of 0.5 gallons of lube oil. Therefore, this is acceptable from a combustible loading standpoint.

This activity is treated as a housekeeping measure. For this activity, the sealant and box enclosure are used as a gasket material to stop the leak and not replace the pressure boundary. This is not a code repair. Measures are in place to prevent the compression fitting from seeing temperatures above 450°F.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

12. EVALUATION: TMOD-WBN-0-2014-044-004 Rev. 2, Evaluation R0

TMOD WBN-0-2014-044-004 is being installed to utilize the Building Heating (BH) system (044) to take the warm water from the common BH system discharge piping and apply to one side of a new temporary plate heat exchanger and then return the flow to the Common Building Heating Circulating Water Pump suction (0-PMP-44-0020 or 0-PMP-44-0017). The secondary side of the plate heat exchanger will contain condensate from the Condensate Storage Tank "B" which will remove heat from the hot side (BH) and warm the condensate. The warm condensate will be used for the Steam Generator Secondary Side Hydro and flushing of the Feedwater, Auxiliary Feedwater and Steam Generator Blowdown system. Westinghouse has specified the steam generators should be between 120 and 150°F when the secondary side hydro is performed.

The TMOD will require the use of both trains of the BH system to allow the system to provide plant heating as well as heating of the Unit 2 condensate for the purpose of secondary hydro. The TMOD resulted in a reduction in the BH system heat exchanger outlet temperature to 190° rather than the normal temperature of 240°F which results in reduced heating capacity. There will also be a change to the way the system is operated because the "auto" selection on the main control board prevents the simultaneous operation of both BH Circulating Water Pumps as both will be required. The operation of both pumps results in a new failure mode due to the lack of a backup pump as both pumps are required for operation of the system. The BH system is a non-safety related system, however, the extended unavailability of the BH system with extremely low ambient temperatures could potentially impact components important to safety. The plant will initiate a contingency associated with the TMOD where the additional heat being removed from the BH system for the purpose of heating the secondary side hydro water will be secured if either the ambient temperature is expected to reach 20°F or the building temperature becomes uncontrollable with the additional load. This will return the system to its normal design condition.

Summary of Evaluation:

The Evaluation concludes there is no need to obtain a license amendment in order to temporarily use the BH system for heat-up of the Unit 2 condensate for the purpose of

secondary side hydro. The continued monitoring of Compensatory actions will be taken if necessary to ensure temperatures remain within limits.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

13. EVALUATION: TMOD-WBN-0-2015-030-001, Evaluation R0

The objective of this TMOD) is to temporarily include the Unit 2 Primary Containment and Unit 2 Annulus into the ABSCE boundary to allow for the testing of the Unit 2 Fuel Handling System, and specifically the testing of the Unit 2 fuel transfer system in the transfer tube and refueling canal to be completed. The testing of the fuel transfer system is required to support the initial Unit 2 fuel load after the issuance of the Unit 2 license by the NRC.

Currently, the interim ABSCE boundary supporting WBN Unit 2 construction does not include Unit 2 primary containment, Unit 2 annulus, Auxiliary Building rooms 713-A21and 757-A14. This interim ABSCE boundary was established by DCN 52283, and is planned to be permanently reversed to its original configuration by DCN 55050.

This temporary boundary is similar to the ABSCE boundary that existed prior to 2008 when the temporary construction openings for Unit 2 were established by DCN 52283 and 55050. The final operational boundary will be established by Stage 3 of DCN 55050.

Summary of Evaluation:

The 50.59 screening for Temporary Modification WBN-0-2015-030-001 identified a potential adverse condition related to the capability of the ABGTS system to perform when the temporary modification relocated the ABSCE boundary. In review of associated documentation regarding the safety functions of the ABGTS it was determined that there is adequate filtration and flow capacity of the ABGTS system to meet the design basis requirements.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.

14. EVALUATION: WBN Unit 1 Cycle 14 Core Operating Limits Report (COLR) R0

Input parameters for the hot zero power steamline break mass and energy release analysis (HZP SLB M&E) are conservatively selected on the basis of values calculated for the WBN core. The reload safety analysis for W1C14 required a change to the assumed stuck-rod moderator density coefficient for the HZP SLB M&E analysis in order to support confirmation that the W1C14 reload core design is bounded by the safety analysis of record. This activity is a change to the HZP SLB M&E analysis of record (AOR). The stuck-rod moderator density coefficient used in the AOR is increased from the current value to a bounding value.

Summary of Evaluation:

A 50.59 evaluation is required because NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation", states that "If the effect of a change is such that existing safety analyses would no longer be bounding and therefore UFSAR safety analyses must be re-run to demonstrate that all required safety functions and design requirements are met, the change is considered to be adverse and must be screened in."

The HZP steamline break case for mass and energy analysis inside containment is reanalyzed for Watts Bar Unit 1 Cycle 14 using an updated input for the stuck-rod moderator density coefficient. The results of the reanalysis showed that all limits continue to be met and that the conclusions in the UFSAR remain valid.

Based on this evaluation, it is concluded that this activity does not meet any of the criteria of 10 CFR 50.59(c)(2), and therefore prior NRC approval is not required to implement this activity.