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

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10 CFR 50.59(d)(2)

U. S. Nuclear Regulatory Commission
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Watts Bar Nuclear Plant, Units 1 and 2
Facility Operating License Nos. NPF-90 and NPF-96
NRC Docket Nos. 50-390 and 50-391

Subject: **Watts Bar Nuclear Plant Units 1 and 2 – 10 CFR 50.59 Summary Report**

Reference: TVA Letter to NRC, "Watts Bar Nuclear Plant Units 1 and 2 – 10 CFR 50.59 Summary Report," dated November 2, 2017 (ML17306A989)

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 the Watts Bar Nuclear Plant (WBN), Units 1 and 2 since the last 10 CFR 50.59 report was submitted on November 2, 2017 (Reference). The evaluations summarized in the enclosure cover the period from May 3, 2017 to October 27, 2018 and demonstrate that the described changes do not meet the criteria for license amendments as defined by 10 CFR 50.59(c)(2). During the development of this report, twelve 10 CFR 50.59 summaries were identified as not having been previously submitted and are included in this report. This issue has been entered into the site corrective action program.

There are no new regulatory commitments in this letter. Should you have questions regarding this submittal, please contact Kim Hulvey, Manager of Watts Bar Site Licensing, at (423) 365-7720.

Respectfully,

Anthony L. Williams IV
Site Vice President
Watts Bar Nuclear Plant

Enclosure:

Watts Bar Nuclear Plant, Units 1 and 2 10 CFR 50.59 Summary Report

cc (Enclosure):

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Enclosure

Watts Bar Nuclear Plant, Units 1 and 2
10 CFR 50.59 Summary Report

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Enclosure

1. Evaluation: DESIGN CHANGE NOTICE (DCN) 59007B, R1

Activity Description:

DCN 59007B is to replace Auxiliary Feedwater (AFW) pressure control valves (PCV) 1-PCV-003-0122 and -0132 with cavitating venturis. The description below describes detailed changes required to perform this design change.

Modifications in the Main and Auxiliary Control Room include:
(Stage 3)

- Pressure differential indicating controllers (PDICs) 1-PDIC-3-122A and -132A, which also provide loop power, will be removed from 1-M-4 and a blanking plate will be installed to cover the opening (1-PDI-3-122A and -132A will remain in 1-M-4).
- Pressure differential indicating controller 1-PDIC-3-122C, which also provides loop power, will be replaced with a pressure differential indicator 1-PDI-3-122C in 1-L-10.
- Pressure differential indicating controller 1-PDIC-3-132C, which also provides loop power, will be replaced with a pressure differential indicator 1-PDI-3-132C in 1-L-10.
- Selector switch 1-XS-3-122 will be removed in 1-L-11A.
- Selector switch 1-XS-3-132 will be removed in 1-L-11 B.

Modifications in the Auxiliary Control Room Include:
A Train (Stage 1):

- Cables will be disconnected and/or termination points will be moved to accommodate new instrument loop in 1-L-11A
- A spare output on an existing GEMAC power supply in 1-L-11A will be renamed "1-PX-3-122C" and used to power the reconfigured loop.

B Train (Stage 2):

- Cables will be disconnected and/or termination points will be moved to accommodate new instrument loop in 1-L-11B.
- A spare output on an existing GEMAC power supply in 1-L-11 B will be renamed "1-PX-3-132C" and used to power the reconfigured loop.

Modifications in the Auxiliary Building and Auxiliary Instrument Room Include:

A Train (Stage 1):

- PCV 1-PCV-3-122 and its associated air tubing and airset will be removed.
- Pressure modulator (PM) 1-PM-3-122 will be removed from local panel 1-L-214 along with associated air tubing, airset, and panel isolation valves.
- Associated cables will be disconnected, as required, in 1-L-214.
- Station air and nitrogen supply tubing for 1-PCV-3-122 will be reworked as required and supply valves will be isolated.
- Pump recirculation line will be rerouted to accommodate the installation of the cavitating venturi 1-CAVV-003-122.
- A spare output on an existing GEMAC power supply in 1-R-127 will be renamed "1-PX-3-122A" and used to power the reconfigured loop.
- Cables will be disconnected and internal wiring reconfigured in 1-R-127.

B Train (Stage 2)

- PCV 1-PCV-3-132 and its associated air tubing and airset will be removed.
- PM1-PM-3-132 will be removed from local panel 1-L-222 along with associated air tubing, airset, and panel isolation valves.
- Associated cables will be disconnected as required in 1-L-222.
- Station air and nitrogen supply tubing for 1-PCV-3-132 will be reworked as required and supply valves will be isolated.
- Pump recirculation line will be rerouted to accommodate the installation of the cavitating venture 1-CAVV003-132.
- A spare output on an existing GEMAC power supply in 1-R-131 will be renamed "1-PX-3-132A" and used to power the reconfigured loop.
- Cables will be disconnected and internal wiring reconfigured in 1-R-131.

AFW PCVs (1-PCV-003-0122 and -0132) have experienced numerous maintenance and reliability problems. A cavitating venturi will allow adequate AFW flow for the design function of the Motor Driven Auxiliary Feedwater (MDAFW) pumps, but prevent pump runout should an AFW or Main Feedwater (MFW) pipe break occur. At MDAFW minimum required flow conditions (approximately 410 gpm) the venturi is sized not to cavitate, so it operates just like any other orifice. The cavitation mode should only occur if the pump is approaching a "run out condition." The venturi is sized to limit flow from the MDAFW pumps to 650 gpm (plus 30 gpm for recirculation flow leaving 20 gpm margin, runout flow= 700 gpm). In the cavitation mode differential pressure generated from the inlet section to the throat reduces the liquid's absolute pressure to its vapor pressure of the fluid (AFW) and it begins to vaporize or boil. These vapor bubbles begin to physically block the throat passageway, which prevents any additional increase in flowrate. The vapor condenses in the discharge part of the venturi. At this point the fluid has recovered and reenters the piping system at velocities slightly lower than the inlet velocity, reestablishing AFW flow.

Main Steam Safety Valve (MSSV) Set Pressure Change

Updated Final Safety Analysis Report (UFSAR) Section 10.4.9.2 states that the AFW system is designed to deliver water at a pressure ranging from the Residual Heat Removal (RHR) system cut-in point to the lowest MSSV set pressure which is defined as the lowest MSSV setpoint + three percent (3%) tolerance + 3% accumulation.

Westinghouse evaluation SECL-94-066 documented test data for the MSSVs, which prove that the safety valves immediately open to full capacity with essentially no accumulation.

TVA calculation EPM0ED070391calculates the effect of this change on the MDAFW pumps. This change includes the Westinghouse recommended five psi accumulation for conservatism and a two psi pressure drop due to the pressure drop between the MSSV and the inlet to the steam generator (SG). This will lower the maximum pressure that the MDAFW pumps are required to pump against from 1257 psig (currently in the UFSAR) to 1228 psig (i.e., 1185 psig setpoint plus 3% setpoint tolerance+ 5 psi accumulation+ 2 psi pressure drop between the SG and the MSSV inlet).

The UFSAR and the WBN Unit1 Technical Specification Bases are being updated to reflect this new MSSV set pressure.

Summary of Evaluation:

DCN 59007B replaces pressure control valves 1-PCV-003-0122 and -0132 with cavitating venturis. The MSSV set pressure is being revised down from 1257 psig to 1228 psig based on test data and evaluation by Westinghouse. Effects on the UFSAR and Accident analysis were reviewed and it was determined that a license amendment will not be required to implement this change.

2. Evaluation: DCN 59961A, R1

Activity Description:

DCN 59961A will realign Service Air System (SAS) valves and reinstate the Unit 2 SAS to its pre-2008 configuration (Condition Report (CR) 146721, DCN 52309, and DCN 52467). A portion of this activity includes opening locked-closed valves 2-ISV-033-0543 and 2-ISV-033-0509.

The only scope in the subject DCN being addressed by this evaluation is opening locked-closed valves 2-ISV-033-0543 and 2-ISV-033-0509, which screened in because they were discussed in the UFSAR as being required to be locked closed. The remainder of the scope screened out of requiring an evaluation and will not be considered in this evaluation.

Summary of Evaluation:

Opening locked-closed valves 2-ISV-033-0543 and 2-ISV-033-0509 will not impact Unit 1 safety related structures, systems and components (SSCs) by increasing the likelihood of malfunctions, the consequences, or create new malfunctions not addressed by the UFSAR. The valves are non safety related and are located in the turbine building. System 033 does not correspond to any initiator for any accident addressed in the UFSAR, an increase in the consequences of an accident, or create new accident scenarios not addressed by the current UFSAR.

3. Evaluation NUMBER: DCN 60696A, R0

Activity Description:

DCN 60696 replaces the existing Pressurizer Power Operated Relief Valves (PORVs) for WBN Unit 1, 1-PCV-068-0340A and -0334. DCN 60696 will replace the Unit 1 PORVs with an updated valve design by Target Rock. Installation of the replacement valves will require minor piping modifications to the Class G Pressurizer Relief Tank (PRT) side of the valve because the new valves have slightly longer flange to flange dimensions than the current PORVs (i.e., the existing PORVs are approximately 22.75" long compared to 24.25" length for the new PORVs). The new valves have a slightly longer open/closed stroke time. The Cold Overpressure Mitigation System (COMS) analysis has been revised and the Pressurizer PORV setpoints will be changed due to this open/closed stroke time increase and flow coefficient change. The WBN Unit 1 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) included in the WBN System 68 system description document is also revised to document the change in PORV setpoints.

Summary of Evaluation:

The change is acceptable. The COMS analysis has been performed and shows that with the new valves and the revised setpoints for COMS/Low Temperature Overpressure Protection (LTOP) will meet 10 CFR 50 Appendix G requirements.

4. Evaluation NUMBER: DCN 62151A, R3

Activity Description:

Stage 1

This test will determine and set Essential Raw Cooling Water (ERCW) throttle valve positions to meet the flow requirements specified in the Test Scoping Document (TSD) issued with DCN 62151A. These valve positions are set to ensure continued reliable operation of Unit 1 only while also preparing the system to be flow balanced in a dual unit configuration in the future. Due to the unit shared design of ERCW, these flow rates are set during Unit 1 only operation such that the valve positions are predicted to maintain the flow requirements for Unit 1 during and following the dual unit ERCW flow balance performed in support of Unit 2 operation. This DCN and associated TSD are issued as a method of minimizing risk to the highest degree possible.

Revision 1 of this evaluation includes the reduction of ERCW flow to the Electric Board Room (EBR) and Main Control Room (MCR) condensers associated with the chillers, provides for the option to open the crossties downstream of each pair of ERCW strainers, and a requirement for ERCW Pumps to perform at 93% or greater of factory acceptance testing for dual unit operation.

Stage 2

Revision 2 of this evaluation includes the establishment of the dual unit ERCW flow rates to the Component Cooling Water System (CCS) heat exchangers (HX) as listed in calculation EPM-JFL-120285. This change includes setting of the intermediate valve positions for valves 1/2-FCV-067-0146-A and 0-FCV-067-0152-B and removal of the 1000 gpm flow restriction for CCS HX "B."

The ERCW outlet and bypass control valves for CCS HX "B" will now be in an active configuration, corresponding to the capability for remote manual operation of the valves from the control room, including during postulated post Loss of Coolant Accident (LOCA) recirculation phase operations. The active configuration of these valves is consistent with existing design for the corresponding CCS HX "A" ERCW control valves and the emergency operating procedures.

Stage 3

During the specific case where one unit experiences a Design Basis Event (DBE) or a LOCA plus Loss of Offsite Power (LOOP) plus loss of a single power train and the other Unit is on RHR within 48hrs after its shutdown, a third ERCW pump will be required to support the proper cooling of both Units. New manual actions are generated as a result. Although this change is not required until Unit 2 fuel load, the impacts to Unit 1 will be included in this DCN. New Technical Specifications sections are created in the License Amendment Request contained in letter CNL-15-088 and are required for Unit 1 to support operation of Unit 2. Alternatively, the

shutdown unit could be cooled via the steam generators for the first 48 hours of shutdown. This evaluation will focus on the situations where a third pump is required.

Summary of Evaluation:

The evaluation considers the impact caused by requiring the normally isolated trained Containment Spray HX being opened to ERCW flow while the associated train is being tested. Opening ERCW flow to a Containment Spray HX 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 Containment Spray HX at the initiation of any non-normal event or accident.

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

The evaluation also considers the impact caused by reducing the capacity of the EBR and MCR chillers. These chillers have the ERCW flow to the condensers reduced, which reduces their capacity. The heat loads in the rooms served are less than the capable cooling capacity of the chillers and the chillers continue to serve their design function.

Dual unit alignment on ERCW may lead to excessive differential pressure across the 2A and 2B ERCW strainers depending on the conditions. The suction of the 2A strainer is in parallel with the 1A strainer and the suction of the 2B strainer is in parallel with the 1B strainer. This change allows for the strainer crossties to be opened between each pair of strainers so that each pair of strainers can share the high load on the 2A and 2B ERCW headers. Opening these strainer crossties is expected to cause slight increase in differential pressure downstream of the strainers due to additional fittings in the flow path, but this differential pressure has been modeled and is shown to cause a negligible impact on the system flow balance. Dual unit ERCW flow balanced conditions and failing of non-accident, non-seismic inside containment chillers results in ERCW Pumps that are required to maintain at least 93% performance in comparison to their factory acceptance curves for dual unit operation. These pumps currently operate at a higher performance and future inservice testing will ensure that the pumps meet this criteria.

Stage 2

The establishment of the dual unit ERCW flow rates to the CCS HX ensures that all three CCS HX receive adequate ERCW flow during all Unit 1 operating alignments. This change has been prior to operation of Unit 2; therefore this evaluation will only focus on the impacts to the operating unit. The completion of the dual unit flow balance has confirmed that all of the equipment serviced by ERCW will receive its design basis minimum ERCW flow rates during all modes of operation (after reductions made in this DCN to the MCR and EBR chillers and CCS HXs). The ERCW outlet and bypass control valves for CCS HX "B" will now be in an active configuration, corresponding to the capability for remote manual operation of the valves from the control room, including during postulated post-LOCA recirculation phase operations.

Stage 3

Requiring three ERCW pumps when one Unit experiences a DBE while the other Unit is on RHR cooling during the first 48 hours after shutdown, ensures that all equipment will receive

their required flow rates during dual unit operation. Additionally, the manual actions required to start a third ERCW pump, including mission dose, are achievable. This stage requires approval of the License Amendment Request contained in letter CNL-15-088 which creates new TS sections to support this change.

All Stages

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, RCS changes, or changes to primary containment design pressure. This change is not the initiator of any new accident nor does it result in a malfunction with a different result. There is not more than a minimal increase in a malfunction, likelihood or consequences of an accident and there is not a change in evaluation methodology.

The conclusion is that Stages 1 and 2 of the proposed change can be implemented per existing processes without obtaining a licensing amendment. Stage 3 will require approval of the License Amendment Request contained in letter CNL-15-088 due to the generation of new TS sections to support this change.

5. Evaluation NUMBER: DCN 64013A, R2

Activity Description:

Removing the flow limitation for CCS Pumps (CCSP) 1A-A, 1B-B, C-S, and 2B-B for the following cases. For CCS Unit 1 Cold Shutdown, Unit 2 LOCA Recirculation Train 1A (requires 8617gpm) and for CCS Unit 1 Refueling, Unit 2 Hot Shutdown Train 2A (requires 8650gpm). These new flow rates affect the loads on the Diesel Generator (DG) System. Each CCS pump is rated 350hp. For Unit 1 only operation, CCSP 1A-A, 1B-B, C-S are required to be operating at 360 Brake Horsepower (BHP) (103% of its rating) to meet these flow requirements. For dual unit operation, the BHP requirements for the CCS pumps 1A-A, 1B-B, 2B-B is increased (varying from 106% to 109% of rating) to meet the increased flow requirements. Requirement from CCSP C-S is increased to 106% when associated with train B (DG 2B-B); but, it is reduced to 96% when associated with train A (DG 1A-A). Calculation EPMGDU031093 recalculates the BHP of CCSP based on the revised flow requirements.

During the specific case where one unit experiences a DBE and the other unit is on RHR within 48hrs after its shutdown, a second CCSP on Train B will required to support the proper cooling of both units. Although this change is not required until Unit 2 fuel load, the impacts to Unit 1 will be included in this DCN. Alternatively, the shutdown unit could be cooled via the SGs for the first 48 hours of shutdown. This evaluation will focus on the situations where a second pump is required.

This change allows for isolation of a Train B section of a single CCS Surge Tank such that a single Train B section of a CCS Surge Tank provides the inventory control for both Unit 1 and Unit 2 CCS Train B. This change is required due to the movement of water volume between the CCS Surge Tanks when adjusting CCS Train B flow using valves 1, 2-FCV-70-153-B with both Train B sections of each surge tank are in service.

Summary of Evaluation:

The new BHP impacts the loading on all DGs based on the analysis performed in the DG loading calculation EDQ00099920080014 for dual unit operation. Listed below are the load changes for the flow rate changes associated with DCN 64013.

Diesel	Load Change	Margin Change
DG 1A-A	Reduce in loading by 3.9kW/4.7kVA	Increase in margin from 6.5% to 6.6%
DG 1B-B	Increase in loading by 11.3kW/10.3kVA	Decrease in margin from 15.3% to 15.1 %
DG 2A-A	Reduce loading by 5.7kW/5.24kVA	Increase in margin from 9.2% to 9.3%
DG 2B-B	Increase in loading by 16.8kW/15.4kVA	Decrease in margin from 12.9% to 12.4%

Therefore, higher BHP requirement from CCSPs impacts the loading on the associated DG. Per the DG Loading analysis in calculation EDQ00099920080014, the worst case steady state loading is on DG 1A-A. The combined impact of revised BHP requirements for CCSP 1A-A and CS, both powered from DG 1A, results in slight reduction in the steady state loading on DG 1A-A. The worst case impact of revised BHP requirements is on DG 2B-B with both CCSP 2B-B and C-S operating at 106%. However, DG 2B-B is lightly loaded compared to DG 1A-A; therefore, the margins calculated in the DG loading analysis calculation is based on the worst loading on DG 1A-A and are not adversely affected by the changes in DCN 64013.

With the above described change the design function of the DGs is not adversely affected.

Requiring two pumps on CCS Train B for certain, limiting conditions, ensures that all equipment receives their design basis flowrates to support the proper cooling of both the accident unit and the shutdown unit. This change does not result in more than a minimum increase in frequency or consequences of an accident or malfunction, does not create a new accident or malfunction with different results, has no impact to fission product barriers and does not use a different methodology as described in the UFSAR.

Allowing the isolation of a Train B Section of a CCS Surge Tank is evaluated to ensure that all equipment functions are served during normal and abnormal conditions, including the required additional manual operator action to establish ERCW emergency makeup to the CCS Surge Tanks. A single Train B Section of a CCS Surge Tank provides the required expansion and contraction capability while also providing a volume of water which is used to detect system leaks. Actions were previously required to establish ERCW emergency makeup to the CCS Surge Tanks and this change adds an action to open one additional manual valve in the immediate vicinity. The loss of the alternate passive vacuum breaker on the opposite surge tank is shown to have no impact for credible bounding events and is of reliable construction such that a mechanical failure which would prevent valve function is not credible. Based on the evaluation, this change does not result in more than a minimum increase in frequency or consequences of an accident or malfunction, does not create a new accident or malfunction with different results, has no impact to fission product barriers and does not use a different methodology as described in the UFSAR.

6. Evaluation: DCN 64326A, R0

Activity Description;

The existing analog Turbine Driven Auxiliary Feedwater (TDAFW) Governor Control System is being upgraded to a digital system in response to industry obsolescence issues associated with the end of life and lack of continuing vendor support of the existing Woodward EG-type turbine controller. Because of the widespread use of digital technology in the non-nuclear industry, a suitable analog replacement for the existing system is not available. This modification will replace the TDAFW turbine controls with a new, industry standard digital control system similar to that being used on reactor feed pump and auxiliary feed pump control systems in the nuclear industry.

This modification has been implemented on WBN U1 by Engineering Change Package (ECP) 58314A.

Description:

The existing WBN Unit 2 TDAFW speed governor system will be replaced in its entirety by this modification. The replacement governor system will replicate the functionality of the existing speed control and electrical overspeed trip components of the existing system, as well as incorporating the flow control function of the existing Foxboro Spec 200 components located in panel 2-L-381A. Incorporating the flow controls functions within the new governor system will simplify the control system and reduce the number of active components required to perform the TDAFW design function, as well as allowing the use of a standard replacement governor design across the TVA nuclear fleet.

The new governor system is configured to accept a 4-20 mA flow signal from 2-FM-46-57D, and will perform the flow control function of the existing 2-F-46-57 Foxboro Spec 200 control loop.

This modification will remove and replace the existing governor panel 2-L-326 in its entirety, and install a new actuator positioner panel 2-L-326A to house the actuator positioner (controller) and associated components. The new 2-L-326 panel will be installed in the same location as the existing panel. The new 2-L-326A panel will be installed near the new 2-L-326 panel.

The existing two magnetic speed pickups 2-SE-46-56 and 2-SE-46-57 on the AFW turbine will be replaced with similar magnetic pickups and vendor supplied cables to the TDAFW skid mounted junction box. Existing speed pickup wiring from the TDAFW skid mounted TB3 to panel 2-L-326 will be re-used.

Existing TDAFW MCR speed indicator 2-SI-46-56A and speed setpoint indicator 2-XI-46-54A on panel 2-M-4 will be replaced with new indicators to accept a 4-20mA signal from the replacement system. The indicated parameters, scaling, and labeling are unchanged.

Existing TDAFW MCR pump demand indicator 2-XI-46-57 on panel 2-M-4 will be replaced with a new TDAFW flow setpoint indicator. This indicator will now display the governor flow setpoint, and will be labeled to reflect the revised displayed parameter. The new 2-XI-46-57 panel meter scale will be similar to existing flow indicator 2-FI-3-142A. A new 4-20mA/4-20mA 1E/non-1E signal isolator 2-XM-46-57 will be installed in panel 2-L-381 to drive this indicator.

Existing TDAFW MCR flow indicator 2-FI-3-142A on panel 2-M-4 and Auxiliary Control Room (ACR) flow indicator 2-FI-3-142C on panel 2-L-10 are unaffected by this modification.

The flow control function of the existing Foxboro Spec 200 AFW turbine flow control components in 2-L-381A will be incorporated into the new TDAFW governor. This will result in the removal of those components used for the flow control function in that rack. The existing Foxboro Spec 200 components in 2-L-381A will be re-configured to maintain the square root flow function, the excess flow switch function, and signal isolation function for remote indication.

The existing Foxboro Spec 200 control stations 2-FIC-46-57B-S in local panel 2-L-381 and 2-FIC-46-57 A-S in MCR panel 2-M-4 will be replaced with standard Electroswitch MW2 modules 2-HC-46-57B-S 2-HC-46-57A-S, respectively.

In MCR panel 2-M-4 the new Electroswitch control switch and module 2-HC-46-57A-S will have Pull-to-Manual operation. A momentary spring return to center Raise and Lower position while pulled will be used to provide for manual control of the new governor speed setpoint. A similar Electroswitch control switch and module 2-HC-46-57B-S will be used in local panel 2-L-381. 2-HC-46-57B-S will allow for adjustment of the governor flow control setpoint while in Auto, while 2-HC-46-57A-S will only allow manual control of governor speed, which replicates the existing functionality of the Foxboro Spec 200 controllers. Both 2-HC-46-57A-S and 2-HC-46-57B-S will incorporate Red and Green indicating lights for Auto and Manual status indication, similar to the existing control stations.

A new interposing relay 2-RLY-46-57A will be added in panel 2-L-381 to provide 1E/non-1E isolation of transfer switch 2-XS-46-57A from the annunciator circuit for annunciator panel 6F window 148D. This is consistent with the existing isolation methodology used for 2-XS-46-57 which also inputs to annunciator panel 6F window 148D.

Panel modifications to local panel 2-L-381 compartment C, and the MCR 2-M-4 panel will be Required to mount the new control switches. Existing TDAFW idle potentiometer 2-XC-46-54 located in local panel 2-L-381 will be removed. The idle set and overspeed test functionality is incorporated into the new system. The existing TDAFW high flow annunciator window in local panel 2-L-381 compartment C is unaffected by this modification and will remain.

The existing servo amplifier and vertically mounted hydraulic actuator 2-SM-46-57B, associated oil tubing, and linkage assembly on governor valve 2-FCV-1-52 will be removed. The replacement actuator will be a horizontally mounted electrical linear actuator 2-MVOP-1-52. The installation of this electrical actuator will require replacement of the existing governor valve stem, the installation of an actuator mounting bracket on the governor valve bonnet. The existing governor bonnet will require machining to accept the new electrical actuator and spring. TDAFW oil system tubing associated with the servo amplifier and hydraulic actuator will be removed, and a blanking plate installed in place of the servo amplifier with a short (approximately six inches) tubing jumper to the existing attached oil sump. The turbine governor valve leakoff line will require re-configuration at the attachment to the valve to avoid interference with the new actuator mounting bracket.

The electronic over speed trip function provided by tachometer 2-SM-46-56 is replaced by an overspeed trip output from the Woodward 505 governor. The existing electronic overspeed trip annunciator in the MCR will be relabeled and repurposed to alarm on either an electronic overspeed trip or a generic TDAFW control system trouble alarm. The existing mechanical overspeed trip and associated annunciator are unaffected by this modification.

The existing function to transfer the TDAFW flow control loop to Auto (flow control) on either an

Accident Signal or TDAFW pump high flow is maintained with the replacement system. The existing close-to-actuate parallel contact configuration will be modified to be an open-to-actuate series contact configuration, which will provide the same logical function. Existing spare contacts/connections on the existing flow switch 2-FS-46-57 and accident relay 2-RLY-46-R2-S will be used for this function.

The existing Spec 200 flow control logic will maintain the flow loop in automatic control upon receipt of a momentary high flow or accident signal, and continue to control flow in automatic at the rated setpoint. The revised logic will return the control loop to manual when the accident signal is reset or flow drops below the switch reset value if any local or the Main Control Room control switch is in the manual position. This difference does not adversely impact the existing force-to-auto function with an accident signal present.

The existing 2-FS-46-57E reset value of 1% (13 gpm) will be revised to be 5% (65 gpm) to provide for margin between the high flow setpoint and the nominal return to manual point. This revised logic will continue to provide the required pump runout protection function by ensuring the pump flow and discharge pressure does not exceed the analyzed value in manual control. The increased flow switch deadband will reduce the frequency of manual-to-auto swaps on decreasing SG pressure.

A new Integrated Computer System (ICS) data acquisition device 2-PLC-261-5465, as well as an associated 24VDC power supply 2-PX-261-5465, and fuses 2-FU2-261-5465/1 and 2-FU2-261-5465/2 will be installed in existing panel 0-JB-292-5465 located at coordinates A8-T on elevation 692 of the Auxiliary building. This new data acquisition device will be connected to existing ICS network switch 0-XS-261-5042 located in panel 0-JB-292-5465.

New class 1E signal isolators will be installed in panel 2-L-381 to provide for 1E/non-1E isolation of analog signals from the replacement system to the integrated computer system. These signals will be connected to the new ICS data acquisition device 2-PLC-261-5465 in panel 0-JB-292-5465.

Spare relay contacts will be utilized to provide for digital status signals from the replacement system to the integrated computer system. Coil-to-contact isolation will be used to provide non-1E status signals. These signals will also be connected to the new ICS data acquisition device 2-PLC-261-5465.

The new TDAFW control system is provided with local manual control capability at the TDAFW turbine requiring no electrical power (AC or DC) available by manual operation of the new governor valve actuator jacking screw. Although available, this "Black Start" capability is not credited for any design basis events. The governor valve (2-FCV-1 -52) can be manually positioned to control AFW speed with the actuator jacking screw, or the existing "Black Start" methodology of leaving 2-FCV-1-52 in the normal full open state, and controlling AFW turbine speed by manually positioning the trip and throttle valve 2-FCV-1-51 can be utilized.

The new system is powered from the existing 125V DC sources the same as the existing control system. This modification will result in an increase to the 125V DC battery loading for batteries I (normal feed) and II (alternate feed).

A trip and throttle valve (2-FCV-1-51) position signal is provided to the governor controller via 2-ZS-46-57C (LS-6) limit switch. As the 2-FCV-1-51 valve unseats (not full closed) and engages the limit switch, the contact is made up and initiates the TDAFW governor start sequence. The output of this circuit will initiate a controlled startup of the governor control valve (2-FCV-1-52);

thus minimizing the potential for a possible electrical or mechanical overspeed trip. This limit switch signal, in parallel with a greater than100 rpm speed contact from the 505 governor, will be used to enable the new actuator positioner. Enabling the actuator positioner will stroke the governor valve from its normally open/fail state of full open to the position demanded by the governor.

New cabling and raceway will be installed from the replacement 2-L-326 governor panel to adjacent panel 2-L-381, and from panel 2-L-381 to panel 2-L-381A. New cabling and raceway will be installed from the replacement 2-L-326 governor panel to the new 2-L-326A positioner panel. New cabling and raceway will be installed from the new 2-L-326A positioner panel to the new governor valve actuator on the TDAFW skid. New cabling and raceway will be installed from the 2-L-381 panel to ICS panel 0-JB-292-5465 located at A8-T on elevation 692' of the Auxiliary Building.

The new TDAFW control panel and associated components are installed in a similar location as the existing TDAFW equipment which ensures that the separation requirements between the TDAFW and MDAFW systems are maintained per the station's design bases documented in the UFSAR and design criteria. The cables routed between the TDAFW skid and the new TDAFW control panels are routed in existing and new conduits between the two end points. This also ensures the new control panel remains in the same fire zone to maintain separation required for a 10 CFR 50 Appendix R fire event.

Combining multiple functions into a single digital device:

Various individual analog components are being replaced with a single digital device that combines the existing functions into a single digital controller. The new digital controller increases the complexity of the governor and creates the potential for different failure modes.

Summary of Evaluation:

This evaluation has determined that the TDAFW system will continue to meet its design and licensing bases requirements following the implementation of the proposed modification that converts to a digital governor control method for the system.

Because the new TDAFW System components are more reliable that the existing components and no new system level failure mode effects are introduced, the proposed modification does not result in more than a minimum increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The new equipment being installed will not initiate any new system malfunctions. Credit is taken for the TDAFW System for the successful mitigation of the following transients, special events, and accidents. The AFW system supplies, in the event of a loss of the main feedwater supply, sufficient feedwater to the SGs to remove primary system stored and residual core energy. It may also be required in some other circumstances such as the evacuation of the MCR, cooldown after a LOCA for a small break, maintaining a water head in the steam generators following a LOCA, a flood above plant grade, Anticipated Transient Without Scram (ATWS) event, and 10 CFR 50 Appendix R fires. The TDAFW System will not adversely impact any of the systems that have a dynamic interface with the TDAFW System. Namely: Condensate Storage Tanks; ERCW; Main Steam; Feedwater; and 125 Volt DC Power Systems. Therefore, the modification does not result in more than a minimum increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Performance requirements associated with core cooling are unaltered such that fuel integrity will be maintained and the UFSAR analysis of radiological consequences remains bounding. The TDAFW System ability to mitigate any postulated design basis accidents will not be decreased. The new equipment will not initiate any new accidents. The modification will not impair or prevent the ECCS from mitigating the consequences of any design basis accidents. Therefore, this activity does not result in more than a minimum increase the consequence of an accident previously evaluated in the UFSAR.

Failure or malfunction of the new equipment will not prevent or affect the ability of safety related systems or systems important to safety to respond to the accidents describe in the UFSAR. Therefore, implementation of the proposed modification does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR will not be increased. The potential malfunctions of the modified equipment are bounded at a system level in the UFSAR. Therefore, the possibility for an unanalyzed malfunction of an SSC important to safety or an accident of a different type than any previously evaluated in the UFSAR are not created.

As described in the UFSAR accident analysis, no malfunction of the AFW System can cause a transient sufficient to damage the fuel barrier or exceed the nuclear limits as required by the safety design basis. As described in the UFSAR Failure Mode and Effects Analysis (FMEA) Tables 10.4-3 and 10.4-4, no new failure modes are created by replacement of TDAFW governor valve controller. The proposed modification does not adversely impact the technical attributes supporting this conclusion. Therefore, the possibility for an unanalyzed malfunction of an SSC important to safety or an accident of a different type than any previously evaluated in the UFSAR are not created.

The new digital equipment does not necessitate a revision or replacement of any currently used evaluation methodology for the TDAFW System. The modification does not result in a departure from the method of evaluation described in the UFSAR in establishing the design bases or in the safety analyses.

Guidance for evaluation of digital upgrades is contained in NEI 01.01/EPRI TR-102348, Guideline on Licensing Digital Upgrades, Revision 1, March 2002, (NPG Electrical Engineering Design Guide DG-E18.1.25, Digital System Development, Procurement and Implementation require that the supplemental questions from NEI 01-01 be addressed within a conceptual design.) NEI discusses the use of an FMEA and in accordance with the graded approach referenced in NEI 01.01, a detailed component level failure modes and effect analysis was developed to identify those malfunctions important to safety which could be caused by the digital controls.

NRC Information Notice (IN) 2010-10 discusses a digital modification to the LaSalle non-safety-related control rod drive system. The NRC noted that LaSalle's application of 10 CFR 50.59 did not answer all the questions in Appendix A of NEI 01.01 in the associated 50.59 Evaluation. In the Criteria Evaluation below, the questions in Appendix A of NEI 01.01 are provided in italics and answered for the proposed digital upgrade. Following the NEI 01.01 questions the criterion question is answered for the analyzed activity.

This evaluation concludes that implementation of the modification does not require a Technical Specification change, does not require a License Amendment, and therefore may proceed without prior NRC approval.

7. Evaluation: DCN 65062A, R0

Activity Description:

The proposed implementation of this DCN will increase the overall reliability of the Main Steam Valve Vault Ventilation System (MSVVVS) by changing the exhaust fan damper failure mode from "fail closed" to "fail open." During hot ambient conditions, a postulated failure of the exhaust damper controls will not result in the dampers failing to a "closed position" and causing temperatures in the valve vaults to exceed their abnormal maximum temperatures. The MSVVVS is not a safety related system; however, high temperatures are a concern for decreasing the overall qualified life for Environmentally Qualified (EQ) components located in the Main Steam Valve Vaults and the setpoints of the Main Steam Safety Valves (MSSVs) can be affected by low ambient temperatures (< 50°F). The automatic control (modulation) of the exhaust dampers by a wall mounted thermostat is being deleted and the dampers will be manually positioned (open or closed) based on ambient conditions and plant modes. During low temperature conditions, the exhaust fans are automatically shutdown. Door and ventilation covers are installed when the temperatures are below 35°F and electric heating may be provided, if required. Technical Requirement (TR) 3.7.5 requires for periodic monitoring of nominal temperature limits in the vicinity of the MSSVs when they are required to be operable. The TR allows for lower than the normal limit for a maximum of eight hours. For the MSSV(s), the normal and abnormal temperature limit is ≥ 50°F. The automatic control function of the exhaust dampers was not evaluated by the original UFSAR listed FMEA. Based on the items above, changing the dampers from "automatic" to "manual' will not increase the likelihood of a malfunction of an SSC important to safety.

Summary of Evaluation:

Proper operation of the MSVVVS is not a safety-related function and is not required for the mitigation of any UFSAR Chapter 6 or 15 accidents. The design basis function of the MSSVs is not adversely impacted by this proposed design change. Because this proposed change will increase the overall reliability of the MSVVVS by changing the exhaust fan damper failure mode from fail closed to fail open, dampers will not fail close causing the valve vault temperatures to exceed the abnormal maximum temperatures, and the qualified life for EQ of the equipment in the rooms will not be impacted. Actions will be required to control the damper positions manually for the exhaust dampers (open or closed) in response to ambient temperature conditions. Therefore, this change does not result in any new accidents or malfunctions, and does not result in increased frequency or consequences of accidents or malfunctions evaluated in the UFSAR. In addition, no fission product barriers are challenged by this change.

8. Evaluation: DCN 65847A, R2

Activity Description:

DCN 65847 replaces obsolete Unit 1 Ronan annunciator equipment in the MCR with up-to-date Ronan equipment. The currently installed Unit 1 annunciator system was installed under contract 75364-A (Job number 16-4336) and provided by Ronan Engineering. This equipment is no longer supported by Ronan and the system shall be upgraded to match (as closely as possible) the Unit 2 annunciator system supplied by Ronan under contract 70052.

The annunciator system components replaced by this DCN are part of System 055, Annunciator and Sequential Events Recording." The design function of the annunciator system is to alert and inform the control room operator to the existence of an out-of-tolerance condition that requires an operator action. The alarms and annunciators are referenced in multiple Chapters of the UFSAR, however, the annunciator system is not specifically described. The annunciator system is non-safety-related, does not perform any nuclear safety-related functions and cannot cause a unit trip.

DCN 65847 will be implemented in three stages. These stages have been developed in coordination with TVA, and may be worked in numerical order with overlapping, i.e. early starts, to facilitate replacement of the existing annunciator system. All three stages are required to be completed.

This DCN will not alter the design function of the annunciator system. The only time the annunciator system will not function as the existing annunciator system does is during the installation (interim period) of the updated annunciator system. During the time the old system is being removed and the new system is being installed, the interim system will monitor the critical alarms (including audible) identified by Operations. The interim system may monitor the alarms of the remote multiplexers prior to outage as identified by Operations. All alarm points of the remote multiplexers will be available to monitor but only alarms identified by Operations are included in DCN 65847. It will not be using the annunciator windows as the system does during normal operation. The inputs will be displayed one of the monitors in the MCR. This is a change of how the system operates for the duration of the installation of the upgraded system. After the upgraded system is functional, the system will operate in the normal manner and identical to the existing system. The annunciator system will be qualified to Seismic Category I(L) requirements.

Summary of Evaluation:

DCN 65847 (PIC 100220) involves an upgrade which utilizes an Interim Annunciator System (without exceeding 90 days) to allow critical alarm points to be monitored during the annunciator system upgrade. The primary system impacted by DCN 65847 is System 055, "Annunciator and Sequential Events Recording" at WBN Unit 1. The proposed activity is designed and qualified in accordance with all applicable standards and NRC requirements to accommodate allowing an interim mean to alert and inform the control room operator, to the existence of an out-of-tolerance condition that requires an operator action.

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.

9. Evaluation: DCN 66251A, R0

Activity Description:

This DCN upgrades the Main Feedwater Regulating Valves (MFRVs) and Main Feedwater Regulating Bypass Valves (MFRBVs) control hardware to improve reliability and eliminate Single Point Vulnerabilities (SPVs). This DCN changes the valve control hardware for the Unit 2 MFRVs and MFRBVs to make the MFRV and MFRBV control hardware more reliable, and reduce the occurrence of SSC malfunctions that could contribute to loss of feedwater or excess heat removal.

As stated in the 10 CFR 50.59 Screening, this DCN does not introduce any adverse changes. However, this DCN is conservatively screened in for evaluation to demonstrate that the effects of a Common Cause Software Failure (CCSF) failure are bounded by existing analysis. Because a digital device is installed, NEI 01-01 guidelines are followed.

Summary of Evaluation:

This DCN upgrades non-safety analog control hardware equipment to digital equipment. The MFRVs and MFRBVs control feedwater flow, a UFSAR design function (UFSAR 10.4.7.2). Feedwater flow affects two UFSAR analyzed accidents, 15.2.8 Loss of Normal Feedwater, and 15.2.10, Excessive Heat Removal Due to Feedwater System Malfunctions. The MFRVs and MFRBVs are also used in the safety related Feedwater Isolation design function (UFSAR 10.4.7.3).

This DCN eliminates SPVs in the MFRV and MFRBV control hardware and upgrades the pneumatic controls to a more reliable digital positioner. Therefore this DCN does not result in more than a minimal increase in the frequency of occurrence of affected accidents or the likelihood of occurrence of malfunctions of an SSC important to safety. This DCN does not affect the dose consequences of accidents or malfunctions; any fission product barrier design basis limit; or any evaluation method described in the UFSAR for establishing the design bases or in the safety analyses.

The existing failure modes of the feedwater regulating valves are not affected, no new valve failure modes are created, and no new result of the valve failure modes is created.

Thus, this design can be implemented without prior NRC concurrence.

10. Evaluation: DCN 66306A, R1

Activity Description:

DCN 66306A installs Category C digital equipment, based on TVA's procedure SS-E18.15.01, "Requirements for Digital Systems (Real-Time Data Acquisition and Control Computer Systems)". DCN 66306 involves a digital upgrade which upgrades the level control for the WBN Unit 1 Heater Drain Tanks (HDTs) and Feedwater Heaters (FWHs) to eliminate and harden SPVs. The field devices (level, flow, and pressure transmitters) for the HDTs and FWHs will provide input to the new local Foxboro I/A DCS remote input/output (I/O) field cabinets. The Foxboro I/A Distributed Control System (DCS) will send an analog control output to each DVC to modulate valve position, discrete contact output signals to control pumps, use Highway Addressable Remote Transducer (HART) communication for valve position feedback and diagnostic information, and output a discrete contact closure for annunciation.

This modification will require several tasks which include:

- Adding level switches for HDT-3 and HDT-7 (software level switches) and removal of existing physical HDT level switches.
- Replacing the existing level system for HDT-7 with a new level system.
- Adding and replacing transmitters/level indicating controllers for input into the Foxboro DCS for the HDTs and FWHs.
- Adding new Foxboro field cabinets to integrate the new input/output (I/O) for the Foxboro DCS.

- Adding relays for the HDT pumps start/stop permissive.
- Upgrading the existing Foxboro DCS servers and MCR equipment to utilize a virtualization environment (V9.2).
- Adding 15KVA (480:208/120V) transformers to power the Control Processors (CPs) in cabinet 1-R-197 in the Auxiliary Instrument Room (AIR) and the new Foxboro I/O cabinets (1-L-986, 1-L-987, and1-L-988) in the Turbine Building.
- Power the new Foxboro Virtualization cabinet 1-R-197 servers from Unit 2 Preferred Power and U1 TSC Power Boards.

The following components for the HDTs and FWHs are impacted and considered as SPVs:

- Level Indicating Controller (LIC)
- Positioner, Control Air Pressure Regulator
- Level Gage Float Chamber (LG)
- Volume Boosters
- Tubing/Sense Lines
- Level I/P Transducer (LM)
- Level Transmitter (LT)
- Level Switch (LS)
- Power Supply, Relay (Power Transfer)
- Fuse.

Level, flow, and pressure transmitters could potentially fail and cause the loss of indication of that instrument in the DCS. The loss of these devices, and the impact of this loss, are discussed in the FMEA) SPF-WBN-2018-0001.

This DCN eliminates the SPV(s) associated with the following equipment:

- LTs, LICs and LS
  Note: The function of the removed level switches are performed by the DCS software and hardware. Three level transmitters for each HDT are provided by this DCN eliminating the existing SPV associated with the LTs, LICs, and LSs. For HDT-3 in particular level switches 1-LS-6-26C/D and 1-LS-6-26D/C receive a signal from the same transmitter 1-LT-6-26 and provide logic for "Runback Turbogen to 85%" per SDD-N3-6-4002. The use of multiple transmitters and level switches for process signals removes single failure vulnerability and improves the reliability and robustness of the control system. The installation of DCS level control of the HDTs and FWHs eliminates the SPV associated with the level transmitters, level indicating controllers, and switches.
- LIC and LM
  Note: The functionality of the LIC is being provided by the DCS. The installation of the DVCs for the normal and bypass valves for HDTs and FWHs provides significant hardening for each control loop. HDT-3 currently has digital controllers which are replaced by DCN 66306. HDT-7 and the FWHs have pneumatic level controllers which are comprised of several component SPVs including the bellows, input connections, output connections, supply air, internal tubing, nozzles, mechanical linkages, and other components. The replacement of the pneumatic level controllers and positioners with DVCs does not eliminate the SPVs; however, DVCs have fewer components and are more reliable.

Local Foxboro I/A DCS remote I/O field cabinets are installed in the Turbine Building on EL. 729.0'. The new cabinets will contain the necessary field bus modules (FBMs) and field

communications modules (FCMs) for accommodating the process level control of HDTs and FWHs and will interface with the field control processors (FCPs) and Master DCS located in the AIR. As appropriate, it is configured to address SPV design issues ensuring that a loss of a single controller, transmitter, power supply, etc., will not result in a system failure, which could cause a plant trip or runback.

The existing Foxboro DCS consists of two operator stations in the Unit 1 MCR, one engineering station in the Unit 1 AIR, and multiple field I/O cabinets. The existing workstation computers supporting the operator and engineering stations are replaced with redundant Foxboro virtualization servers. The new virtualization servers will use thin clients to serve as the operator and engineering stations. The existing Foxboro I/O field cabinets contain FBMs, FCPs, and FCMs to accommodate local process control and interface with the Foxboro DCS in the AIR. These will require firmware updates in order to communicate with the upgraded Foxboro DCS.

Monitoring and control of the HDTs and FWHs are performed in the MCR via the operator stations. The engineering workstation is in the AIR for FCP programming and configuration. KVM (keyboard, video, mouse) stations local to cabinet 1-R-197 will provide interface for configuration and programming of the virtual machines in the DCS.

Summary of Evaluation:

The DCS system replaces existing analog controls in the existing balance of plant (BOP) systems with DCS and reduces many single point failure vulnerabilities with reliability improvements. The impact of SPVs are reduced by eliminating volume boosters and pneumatic connections. The impact of the SPVs remaining are reduced by detection and annunciation of faults and failures in the DCS and Digital Valve Controllers (DVCs). The new system provides redundant inputs, redundant processors, networks, power supplies with backup power, etc. The new system is designated as Quality Related and is designed to meet Quality Related requirements; the reliability of new system is superior to the old analog system. The modification does not negatively impact any SSC that is important to safety nor does it impact the consequences or the frequency of their occurrence. The upgraded DCS does not cause a new type of malfunction or accident to be created. The upgraded DCS reduces the likelihood of failures and their consequences by providing more reliable and redundant control system. In addition, this modification provides the capability to reduce manual operator actions, thereby, allowing greater opportunity for assessment, monitoring, and response.

The upgrade to DCS and replacement systems results in overall improvement in the plant and the ability to function with individual devices out of service as:

- DCS provides for use of additional input signals for control. The DCS will continue to maintain function with the loss of a single input for critical controls with multiple inputs. In the case of a single input, the last good value prior to the failure is used. The DCS will provide an alarm on the DCS Visual Display Stations (VDUs) for loss of an input.
- The DCS is powered from redundant power sources thus for loss of any single power supply or power source the DCS will continue to maintain control.
- The signal output to plant control devices, such as valves, use redundant FBMs such that should one FBM fail the other FBM maintains the control of the device.
- Important functions are separated on a pair of DCS processors for redundancy.

The proposed modifications do not increase the frequency or likelihood of accidents or malfunctions or create a new type of accident. A hardware-related common cause failure (CCF) conclusion of unlikely (not credible) and a software-related CCF conclusion of not unlikely (credible) were used from the technical support work in addition to previously evaluation were performed via Unit 2 installation of the DCS.

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); therefore, obtaining prior NRC approval is not required to implement this activity.

11. Evaluation: DCN 66360A, R1

Activity Description:

DCN 66360 replaces the Containment Upper (1-RE-90-112), Containment Lower (1-RE-90-106) and Auxiliary Building Vent (0-RE-90-101) Radiation Monitors for Unit 1. These ThermoWestronic monitors are changed to remove the difference between Unit 1 and Unit 2. The new monitors will be similar to the existing monitors with a few differences; the new monitors will have a fixed paper filter for particulate monitoring, all channels will be processed by a digital monitor, and each monitor will have two circuit breakers (one for each pump). The monitors will be replaced with units from General Atomics to be similar to the units installed on Unit 2.

This activity will require four stages to implement this modification:

Stage 1 - rework existing conduit supports and install new supports to remove any interferences in preparation for installation of the new radiation monitors.
Stage 2 - replace radiation monitor 1-RE-90-112 and rework associated sensing lines, cables, and conduits.
Stage 3 - replace radiation monitor 1-RE-90-106 and rework associated sensing lines, cables, and conduits.
Stage 4 - replace radiation monitor 0- RE-90-101 and rework associated sensing lines, cables, and conduits.

The design of these monitors will include off-line channels to provide continuous real-time detection of particulate, iodine, and noble gas (PIG) radioactivity (including collection of particulate and radio halogens for on-site analysis). These monitors: Auxiliary Building Vent (0-RE-90-101), Containment Upper (1-RE-90-112), and Containment Lower (1-RE- 90-106)) will detect and measure the airborne radioactivity concentration in the Unit 1 Containment Building upper and lower compartments and perform real-time detection of the particulate and noble gas radioactivity in compliance with 10 CFR 50 Appendix A, GDC 30 and 64. The particulate channels of 1-RE-90-112 and 1-RE-90-106 are designed to respond to a leakage increase of 1 gal/min (3.8 L/min) in one hour or less as required per Regulatory Guide (RG) 1.45.The upper containment monitor and Auxiliary Building Vent monitor, 1-RE-90-112 and 0-RE-90-101 respectively, will also perform real-time detection of radioiodine. 0-RE-90-101 meets the Intent of the guidance of RG 1.21 and RG 1.97 compliant device for Type E Category 2 variable.

The monitors for containment located on EL. 737' of the Auxiliary Building will have the capability to take a sample from either or both of two separate locations in lower and upper containment in order to provide assistance in locating the general area of abnormal leakage.

The upper containment monitor will have the backup capability to monitor the radioactivity in the containment lower compartment atmosphere, and the lower compartment monitor will have the backup capability to monitor the radioactivity in the containment upper compartment atmosphere. The monitors will be capable of continued operation after loss of off-site power. This satisfies the guidance in Section 12.3-12.4 of NUREG-0800.

The Upper and Lower Compartment monitors are safety-related but do not perform any primary safety functions or initiate any control function. They are classified as safety-related due to RG 1.45 requirements stating that nuclear power plants may be operating when an earthquake occurs and may continue to operate after an earthquake ceases and it is prudent to require the leakage detection systems to function under the same conditions. The Upper and Lower Containment radiation monitors will be qualified to Seismic Category I requirements. The Auxiliary Building Vent radiation monitor will be qualified to Seismic Category I(L) requirements.

The proposed activity involves a digital upgrade of Unit 1 radiation monitors. The new radiation monitors are essentially the same to the existing monitors installed in Unit 2 which are designed to perform these basic functions:

- Give warning of a condition which might lead to radioactivity releases that could result in exceeding the limits set forth in 10 CFR 20, 10 CFR 50, and 10 CFR 100.
- Warn plant personnel of increasing radiation levels which might result in a radiation health hazard.
- Rapidly provide information on fuel clad and equipment failures or malfunctions.
- Provide means of radioactive fluid leakage detection.
- Perform monitoring during normal operation and accident conditions (as required).

Summary of Evaluation:

The proposed modifications do not cause more than a minimal increase in the frequency or likelihood of accidents or malfunctions. The design bases for fission product barriers will not be altered or exceeded. No new method of evaluation was used in evaluating the proposed modifications.

A hardware-related CCF conclusion of unlikely and a software-related CCF conclusion of not unlikely were used from the technical support work in addition to previously evaluation were performed via Unit 2 installation of the monitors.

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.

12. Evaluation: DCN 66453A, R0

Activity Description:

This activity adds acceptance criteria for the Unit 1 ECCS allowable gas values on discharge piping. The ECCS is combination of System 62, Chemical and Volume Control System (CVCS), System 63, (Safety Injection System (SIS), and System 74, Residual Heat Removal System (RHRS). Unit 2 is not addressed by this activity, but a similar change will be performed under Design Change Package (DCP) 66454. This activity also updates the UFSAR, Technical Specification Bases, and System Descriptions with current evaluations in complying with

Generic Letter 2008-01 (GL 08-01), Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems. Note that while the Containment Spray System is specified as part of the GL 08-01, no acceptance criteria for allowable gas is being added for the system.

The purpose of GL 08-01 is to ensure each utility has addressed impacts of accumulated gas on ECCS, Decay Heat Removal (which is Residual Heat Removal at WBN), and Containment Spray System. This gas accumulation would include air, nitrogen, hydrogen, water vapor, or any other void that is not filled with liquid water. WBN previously evaluated GL 08-01 impacts, but did not update the upper tier documentation such as the UFSAR to reflect these changes. Currently the UFSAR Section 6.3 (Unit 1) does not address the presence of gas, but it is implied by the Technical Specification (TS), LCO 3.5.2, ECCS - Operating, which requires two ECCS trains be operable during Modes 1-3. Surveillance Requirement (SR) 3.5.2.3 under that TS is to "Verify ECCS piping is full of water." From NEI 09-10, "Revision 1a, Guidelines for Effective Prevention and Management of System Gas Accumulation," Section 9, because the UFSAR does not address gas within the ECCS, the quantity of acceptable gas is assumed to be zero. The issuance of design output of ECCS allowable gas values greater than zero is therefore a non-conservative change in methodology because it results in more margin as described in NEI 96-07 Section 4.3.8.1. In addition a new UFSAR section 6.3.4.3.1 is being added as part of this activity to describe the existing testing of ECCS to vent and quantify the gas within the system. The TS Bases are also being updated with similar words as the UFSAR to describe the evaluation that has been done and that some gas is acceptable.

This activity uses the guidance from the NRC endorsed NEI 09-10 Revision 1a, Guidelines for Effective Prevention and Management of System Gas Accumulation.

The Westinghouse evaluation, WAT-D-12300, Letter Report Documenting the Evaluation Supporting a 30 Second SI Delay Time, evaluated a change in ECCS delay time from that in the Analysis of Record (AOR) for Large Break LOCA (LBLOCA) only. For this accident, the AOR uses a 12-second delay for offsite power available (OPA) and 32 second delay time for loss of offsite power (LOOP). This is shown on UFSAR Table 15.4-25. The evaluation increases the delay times to a 40 second delay for OPA and 45 second delay for LOOP. This increase in time delay was used in the evaluation of flow delays due to gas voids.

Summary of Evaluation:

The only part of the activity that required evaluation was the increase in ECCS time response for LBLOCA. The LBLOCA AOR currently has time delays of 12 seconds for OPA and 32 seconds for LOOP. The change to 40 seconds for OPA and 45 seconds for LOOP was evaluated by Westinghouse as having a negligible impact on the existing LBLOCA analysis because the new delay time is still within the time when most flow is lost through the RCS Break. Also during this time the accumulator flow is significantly higher than other ECCS flow. Westinghouse estimated a PCT impact of 0°F.

As such, the change does not result in an increase in the frequency of an UFSAR-evaluated accident or malfunction, an increase in the consequences of an accident or malfunction, or create a different type of accident or malfunction than previously evaluated in the UFSAR. The 0°F PCT impact is covered by the more specific 10 CFR 50.46 regulation. Because the change is bounded by the existing LBLOCA AOR, there are no fission product barrier design basis limits being exceeded or altered.

13. Evaluation: DCN 66462A, Evaluation R0

Activity Description:
Certain anticipated operational occurrences for pressurized water reactors can cause an unplanned increase in RCS inventory. Depending on the properties of the injected inventory and the response of the automated controls, this could result in fuel damage or RCS overpressurization. One such event is an Inadvertent Operation of the Emergency Core Cooling System (IOECCS). Once started, the ECCS does not automatically stop, and therefore the IOECCS must be mitigated by manual action prior to incurring damage to the reactor coolant pressure boundary (RCPB).

The IOECCS is analyzed so as to prevent overfill of the pressurizer, as pressurizer overfill can result in a loss of coolant via the Pressurizer (PZR) PORVs or safety valves. Therefore, the event ends when the net inventory addition to the RCS is terminated. The limiting scenario is when the RCS is already pressurized and the pressurizer level corresponds to 100% power at the beginning of the event; for this scenario, the charging pumps are the only ECCS pumps providing forward flow. Because the charging pumps are aligned to the RCS via the cold leg injection, or Boron Injection Tank (BIT) path, the event has historically been considered terminated when the operators diagnose that the ECCS actuation was inadvertent and subsequently terminate ECCS flow via isolation of the BIT path. The analysis assumes this action occurs 10 minutes from the start of the event, and demonstrates no overpressurization or relief from the pressurizer relief valves if the event is terminated at this time.

With the BIT path isolated, however, charging flow can continue to reach the RCS via the reactor coolant pump (RCP) seal injection. Although the Safety Injection Termination procedure requires the re-establishment of pressurizer level control after terminating the ECCS flow, this action is not time critical. As a result, the current analysis assumes all injection stops at 10 minutes and doesn't account for seal injection.

Accounting for seal injection in the analysis is not straightforward, as Regulatory Issue Summary (RIS) 2005-29 stated that the regulator would no longer accept IOECCS analyses in which the pressurizer went water solid. Therefore, the event was reanalyzed by Westinghouse, using a more realistic, site-specific ECCS hydraulic model to determine the actual time to pressurizer overfill. From this analysis, it was determined that establishing letdown in 13 minutes would prevent the pressurizer from reaching water-solid conditions.

The activity therefore consists of updating the design/licensing documentation to document the new time critical step in IOECCS mitigation, and replacing the previous IOECCS accident analysis in the UFSAR with documentation corresponding to the new analysis of record.

Summary of Evaluation:

The UFSAR currently declares that an IOECCS event will not result in pressurizing the RCS to more than 110% of its design value, the DNBR will remain above the 95/95 limit for fuel cladding integrity, and the event will not generate a more serious condition without other faults occurring independently. The power and temperature reduction during the transient assures that DNBR occurs at the start of the transient, while operator action assures that the injection will be terminated prior to challenging the RCS pressure limits or escalating the event to LOCA.

Requiring letdown in a specific timeframe in response to an IOECCS does not increase the frequency of the IOECCS, although it does ensure the event does not escalate to a LOCA via safety valve malfunction. The time to establish letdown is reasonable due to the multiple means via which letdown can be achieved, in addition to additional margin built into the time requirement for contingencies. By performing these actions, no malfunction or accident of a different type is created (i.e., no unisolable release from the pressurizer) and the consequences of the accident remain unchanged because all three fission product barriers remain intact and are not altered by the change.

The changes to the IOECCS analysis to support the change do not constitute changes to elements of the analysis methodology.

14. Evaluation: DCN 66465 Rev. 5, Evaluation R0

Activity Description:

The existing Main Turbine Analog Electrohydraulic (AEH) control system and Moisture Separator Reheater (MSR) controls at Watts Bar Unit 1 Nuclear Plants are obsolete and are replaced by a new Digital Electrohydraulic (DEH) control system by DC 66465. The replacement DEHC system utilizes the Woodward MicroNetTM Triple Modular Redundant (TMR) digital control platform. The DEH system incorporates the control, protective, and monitoring functions of the existing main turbine AEH system and MSR controls with enhancements designed to improve fault tolerance.

The scope of DCN 66465 includes the installation of the DEH control hardware and software provided by Siemens Energy. The scope of DC 66465 also includes the installation of a Foxboro DCS Human Machine Interface (HMI) monitor on MCR panel 1-M-2, and HMI software provided by Framatome (formerly Schneider). The DCS network equipment and network software to support the HMI monitor interface is being installed by DCN 66306. DCN 66306 is a co-requisite to this DCN.

The scope of DCN 66465 includes the removal of the Turbine Trip Auto-Stop Header which is replaced with a new four-valve trip block assembly connected to the high pressure Electrohydraulic Control system (EHC) Emergency Trip Header. DC 66471 replaces the existing low oil pressure switches that initiate a reactor trip on low Auto-Stop Header oil pressure (above the P-9 Power Range Neutron Flux Interlock), with new pressure switches connected to the high pressure EHC Header. TS change request 390-WBN-TS-17-04 has been submitted by DCN 66471 to revise the low oil pressure switches trip setpoints in Unit 1 TS Section 3.3.1, Table 3.3.1-1 (ADAMS ML17075A229). TS change request 390-WBN-TS-17-04 was approved by the NRC by Amendment No. 119, dated March 28, 2018 (ADAMS ML18052B347). DCN 66471 is a co-requisite to this DC.

The following elements of the modification are considered adverse and are screened in for further evaluation under 10 CFR 50.59:

Digital Upgrade

The new Woodward MicroNet TMR EHC turbine control and moisture separator reheater control system installed by DC 66465 is a digital control system. This is considered adverse and is screened in for further evaluation under 10 CFR 50.59 due to the following:

a) This modification combines the functions of several previously separate analog components within the digital devices in the new control system and must be considered and evaluated in accordance with NEI 01-01.
   - Separate components that previously provided separate Turbine Automatic and Turbine Manual control functions are combined in the MicroNet System.
   - Separate components that previously provided separate turbine control and protection functions are combined in the MicroNet System
   - The MSR Control System is incorporated into the MicroNet System.

b) The use of a digital Human Machine Interface (HMI) monitor for turbine control, MSR control, testing, and maintenance functions.

c) The new digital controller increases the complexity of the system and creates the potential for different component failures that must be considered and evaluated for impact on the failure modes of the turbine control system as evaluated in the UFSAR.

d) Software development, configuration settings of the new system to TVA digital system requirements and UFSAR described functions are evaluated in accordance with SS-E18.15.01.

Turbine Trip and Overspeed Protection

The main turbine mechanical overspeed trip device, Auto Stop header and Overspeed Protection Circuit (OPC) functions are replaced with the MicroNet initiated overspeed trips, trip block assembly and the digital IOPS ProTech overspeed trip device. This is considered adverse and is screened in for further evaluation under 10 CFR 50.59.

Turbine Trip on Generator Breaker Opening

The initiating breaker position circuit is modified as the energize-to-trip Turbine Trip on Generator Breaker opening (TTGB) relays are deleted and the trip is now initiated by de-energize-to-trip 152Z relays. Because the use of de-energize-to-trip 152Z relays would cause a turbine trip on a loss of the 250V DC power supply to the relays, the following trip permissives are added to the GAP logic:

- Generator MW is < 50 (based on median select voting from 3 MW transducers)
  or
- Turbine speed is >1806 RPM (based on median select voting from 3 speed sensors)

This is considered adverse and is screened in for further evaluation under 10 CFR 50.59.

Generator Breaker Trip After Turbine Trip

A new generator megawatt signal is added to the generator breaker trip logic so that reverse electrical power is also detected prior to tripping the breaker to comply with Westinghouse Operation and Maintenance Memo (OMM) 092 that recommends tripping the generator breaker within one minute to prevent motoring the generator. This is considered adverse and is screened in for further evaluation under 10 CFR 50.59.

Summary of Evaluation:

This evaluation has determined that the EHC Turbine Control and Turbine Protection Systems (TGCPS) will continue to meet its design requirements following the implementation of the proposed modification that converts to a digital control system.

Because the new DEHC System components are more reliable that the existing components and no new system level failure mode effects are introduced, the proposed modification does not result in more than a minimal increase in the frequency of occurrence of an accident or transient previously evaluated in the UFSAR.

The new equipment being installed will not result in any component malfunctions that could increase the potential for a turbine trip or transient. Nor will any malfunction result in an increase in the potential for a required protective function to be performed (tripping the turbine). Therefore, the modification does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Performance requirements associated with core cooling are unaltered such that fuel integrity will be maintained and the UFSAR analysis of radiological consequences remains bounding. The new equipment will not initiate any new accidents. The modification will not impair or prevent the ECCS from mitigating the consequences of any design basis accidents. Therefore, this activity does not result in more than a minimal increase the consequence of an accident previously evaluated in the UFSAR.

Failure or malfunction of the new equipment will not prevent or affect the ability of safety related systems or systems important to safety to respond to the accidents describe in the UFSAR. Therefore, implementation of the proposed modification does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR. The potential malfunctions of the modified equipment are bounded at a system level in the UFSAR. Therefore, the possibility for an unanalyzed malfunction of an SSC important to safety or an accident of a different type than any previously evaluated in the UFSAR is not created.

As described in the UFSAR accident analysis, no malfunction TGCPS can cause a transient sufficient to damage the fuel barrier or exceed the nuclear limits as required by the safety design basis. Therefore, the possibility for an unanalyzed malfunction of an SSC important to safety that can challenge a fuel barrier or an accident of a different type than any previously evaluated in the UFSAR is not created.

The new digital equipment does not necessitate a revision or replacement of any currently used evaluation methodology. The modification does not result in a departure from the method of evaluation described in the UFSAR in establishing the design bases or in the safety analyses.

Guidance for evaluation of digital upgrades is contained in NEI 01-01, Guideline on Licensing Digital Upgrades, March 2002. NRC Information Notice (IN) 2010-10 stated the NRC expectation that all the questions in Appendix A of NEI 01-01 should be considered in a 50.59 Evaluation for systems that could cause a plant trip or reactivity transient. In the Evaluation below, the questions in Appendix A of NEI 01-01 are provided in italics and answered for the proposed digital upgrade.

NEI 01-01 discusses the use of an FMEA. In line with the graded approach referenced in NEI 01-01, an FMEA (20583-FMEA-L583001-05215) was prepared to identify those malfunctions important to safety the digital controls could cause.

The 50.59 screening review concludes that implementation of the modification does not require a TS change. This evaluation concludes that implementation of the modification does not require a License Amendment, and therefore may proceed without NRC approval.

15. Evaluation: DCN 66576 Rev. 1, Evaluation R0

Activity Description:

Unit 2 currently uses the WCAP-10325-P-A methodology as the UFSAR-described analysis to predict the mass and energy releases from a design basis LOCA. The methodology is also used to determine the peak pressure resulting from the accident to assure containment integrity. Several errors were recently discovered in WCAP-10325-P-A that, when corrected, resulted in increasing the calculated peak pressure. These errors could have been offset via the addition of a significant quantity of ice to the ice condenser; however, the analysis also needed to account for the performance of heat removal systems that are shared between Unit 1 and Unit 2.

To meet these needs, Watts Bar Unit 2 will adopt the WCAP-18204-P analysis to support the UFSAR's description and assurances for the containment functional design demonstrating that the existing Unit 1 WCOBRA/TRAC LOCA MandE release and LOTIC1 containment integrity analyses that consider the Model 68AXP replacement steam generators (RSG) is bounding and applicable to Unit 2 with Model D3 SG . WCAP-18204-P is an engineering report that calculates the LOCA mass and energy releases and peak pressure using the WCAP-17721-P-A (WCOBRA/TRAC) methodology, which is a significantly more mechanistic model. WCAP-17721-P -A predicts an acceptable peak pressure with no changes in initial ice mass, even under the consideration of sharing the heat removal systems between two units.

The post-accident temperature and pressure profiles predicted using the WCAP-17721-P-A methodology for the containment harsh environments were not completely bounded by the previous design basis profiles; therefore, the adoption of WCAP-17721-P-A as the LOCA containment integrity design basis results in the creation of new profiles to which safety-related equipment in these harsh environments must be qualified.

Summary of Evaluation:

The design function of primary containment is to assure that an acceptable upper limit of leakage of radioactive material is not exceeded under design basis accident conditions. The design function of the containment sump is to provide a water source for ECCS and containment spray. The use of WCAP-17721-P-A is solely a methodology change and does not add, remove, or modify any SSCs or procedures. It does not constitute a test or experiment, and it poses no impact to the plant's TS. Because the current LOCA mass and energy releases and resulting containment pressure are calculated using a methodology explicitly described in the UFSAR in Section 6.2.1.3, the use of WCAP-17721-P-A is a change in a method of evaluation used to establish the design basis as part of the safety analysis. In addition, it predicts different post-accident temperature profiles for the containment harsh environments that could impact the qualification of safety-related equipment in these spaces.

The NRC generically approved WCAP-17721-P-A methodology for reference in license basis applications for the large-break LOCA mass and energy release calculations in an SER issued August 24, 2015 (ADAMS ML15221A007). The specific use of WCAP-17721-P-A to generate the WCAP-18204-P engineering report for Watts Bar Unit 2 satisfied all the limitations and conditions stipulated by the SER. Therefore, the NRC has approved this change to a new evaluation methodology to be used in the UFSAR to establish the design basis or safety analyses as defined by NEI 96-07 Revision 1, Section 3.4.

The changes to the post-accident parameter profiles incurred by crediting the WCAP-18204-P engineering report does not adversely affect the environmental qualification of any safety-related equipment. As such, it does not result in an increase in the frequency of an UFSAR-evaluated accident or malfunction, an increase in the consequences of an accident or malfunction, or create a different type of accident or malfunction than previously evaluated in the UFSAR. The new methodology demonstrates there are no fission product barrier design basis limits being exceeded or altered.

16. Evaluation: DCN 66577 Rev. 0, Evaluation R0

Activity Description:

This change revises Table 15.5-23 of the UFSAR to update doses due to a Fuel Handling Accident (FHA) for the MCR due to increased response time for the MCR isolation dampers. The change also revises SDD-N3-30CB-4002, Table 9.8, Note 2 to update the total damper isolation time for dampers 0-FCV-031-0003-A and 0-FCV-031-0004-B. This is not a physical change to the plant, rather a documentation change to reflect the updated consequences from using a more conservative response time for the rate meters which cause a Control Room Isolation (CRI) in response to elevated radiation in MCR normal ventilation air intakes.

Design Basis Accidents involved:

The Scope of this change is limited to FHA.

Summary of Evaluation:

The proposed change does not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction beyond the ten percent allowed by 10 CFR 50.59, or create a new type of accident. The design bases for fission product barriers will not be altered or exceeded. No new method of evaluation was used in evaluating the proposed change.

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.

17. Evaluation: TMOD WBN-0-2017-067-002 Rev. 0, Evaluation R0

Activity Description:

The ERCW Pumps on the B-Train will have their associated discharge isolation valves (0-ISV-67-504E, -504F, -504G, and-504H) replaced in 2018. This is required to ensure the valves will provide an adequate boundary for the PMs associated with the B-Train Pumps' check valves (0-CKV-67-503E, -503F, -503G, and -503H). This portion of the ERCW System is common to both units and is required by TS if any unit is in operation. WBN has elected to remove the ERCW B-Train from service to allow for the replacement of the discharge isolation valves. Removing the B-Train of ERCW from service for the maintenance activity will require removing all supporting B-Train equipment from service (securing) to prevent an inadvertent start of equipment without cooling flow to those components. This includes all B-Train ERCW Loads and the B-Train of CCS, ECCS, Diesels, and various B-Train Chillers/Coolers.

One of the concerns related to removing the B-Train from service is the impact on containment temperatures and RCP motor winding temperatures. The Technical Evaluation associated with this TMOD concludes that containment temperatures should be adequate with A-Train Lower Containment Coolers (LCCs) in service (with the LCCs having their respective TCVs wide open) and B-Train LCC Fans in operations with ERCW temperatures <60 °F. However, to ensure there is adequate containment cooling, a temporary supply of cooling flow will be supplied to the B-Train LCCs for both units. While the ERCW B-Train is removed from service, RCW will be supplied via an 8" flood mode spool piece and opening of 0-ISV-67-532B and 0-ISV-24-826. This will provide flow to the ERCW1B Header and will also be supplied to the 2B Header by use of the cross-tie line in the Intake Pumping Station (IPS). This is a compensatory measure utilized to minimize the risk to the plant during the maintenance activity.

Summary of Evaluation:

The Evaluation only evaluates the temporary supply of RCW to the B-Train Lower Containment Coolers of each unit. The B-Train of ERCW will be removed from service and all supportive B-Train Equipment will be rendered inoperable by the maintenance activity. Removing equipment from service (making it inoperable) for maintenance within the TS allowed outage time does not require a 50.59 Evaluation and is addressed by the plants implementation of 10 CFR 50.65(a)(4).

The proposed change does not impact the ability of the LCCs to perform their safety functions of circulating air during any Non-LOCA accident. The proposed change is only changing the source of water provided to the LCCs, CRDM Coolers, and RCP Motor Coolers from ERCW to RCW which is strained to the same requirements and does not exceed the pressure/design temperatures of the ERCW System. The proposed change does not introduce a new malfunction, a new accident, or increase the consequences from a design basis accident. It is concluded that the temporary supply of RCW will support the design function of the LCCs to maintain the lower containment temperatures within the TS limit of 120°.

18. Evaluation: TMOD WBN-1-2017-030-001 Rev. 0, Evaluation R0

Activity Description:

Control Rod Drive Mechanism (CRDM) Cooler 1A-A tripped unexpectedly on July, 9, 2017. Troubleshooting identified a hard ground on 1-MTR-30-83/1 as the cause of the trip. This motor is located inside the Polar Crane Wall and is inaccessible during plant operation and requires an outage to replace. Furthermore, 10 CFR 50 Appendix R requires the CRDM Coolers to be available to perform a required Fire Safe Shutdown (FSSD) function. This motor failure has caused Operations to enter OR 14.10.

T-MOD WBN-1-2017-030-001 will temporarily disable the failed motor of CRDM Cooler 1A-A to allow CRDM Cooler Fan 1A-A Fan #2 to be operated without Fan #1. This will be done by removing the six primary disconnect finger clusters from the rear of 1-BKR-30-83/1 and insulating the line and load side stabs. This will allow the controls of CRDM 1A-A to operate as designed but will prevent 1-MTR-30-83/1 from being energized. Disabling a fan stage will require this cooler to be maintained in Supplemental Lower Compartment Cooling ("bypass") mode, which is the required alignment for an Appendix R event. Implementation of this TMOD will restore FSSD functionality to CRDM Cooler 1A-A and allow Operations to exit OR 14.10.

Enclosure

Summary of Evaluation:

The design function of the CRDM Coolers is to maintain acceptable temperature with the CRDM shroud for the protection of equipment and controls during normal reactor operation and normal shutdown. The CRDM Cooling system is designed to operate with the LCCs to maintain a maximum air temperature with the upper reactor cavity of 120°F and to route all of the reactor well air through the CRDM shroud to maintain a maximum air temperature of 185°F. Air drawn through the CRDM shroud is cooled by the active fan-coil assemblies to approximately 120°F and discharged into the lower compartment of the Reactor Building. When additional cooling in the lower compartment is required, the arrangement of dampers allows either or both standby CRDM fan-coil assemblies to recirculate air in the lower compartment general spaces and supplement the LCC system capacity.

The four CRDM air cooling fan-coil assemblies are located in the main lower compartment space at floor Elevation 702.78. Each assembly consists of a plenum, three air cooling coils, two vaneaxial fans in series, air operated dampers, instruments, and controls. The four CRDM coolers are divided into two pairs. Cooler 1A-A is paired with 1D-B, and 1C-A is paired with 1B-B. One cooler in each pair is required to provide adequate cooling to the CRDM shroud during normal operation. Each fan motor has its own breaker, such that two breakers operate together for each CRDM Cooler. The control circuit for Fan 1 contains all the control logic and operates the breaker for Fan 1. There are no additional controls in the circuit for Fan 2, as its breaker is just "slaved" off the status of the Fan 1 breaker, such that the two breakers open and close together to control the double fan/motor assemblies of each cooler.

The CRDM Coolers and associated duct/dampers are not safety-related and are not required to perform a primary nuclear safety function. However, the CRDM Cooler along with the LCCs are required for safe shutdown, per 10 CFR 50 Appendix R to keep containment temperatures from exceeding operability limits on safe shutdown equipment inside containment. 1-BKR-30-83/1 is safety related to protect 480V Shutdown Board 1A1-A.

This TMOD will disable CRDM Cooler 1A-A fan/motor set #1 by removing the six primary disconnect finger clusters from the rear of 1-BKR-30-83/1 and then insulating the breaker and/or switchgear side line side and load side contacts ("stabs"). These finger clusters are the components that connect the breaker to the stationary line and load side contacts in the switchgear itself. With these removed, there will be a gap between all primary contacts on the breaker and the switchgear. This will allow the breaker for Motor 1 to be racked in and operated without energizing its load (failed Motor 1). The breaker for Motor 2 will continue to follow Breaker 1 and will energize Motor 2 and allow Motor 2 to be operated without energizing Motor 1. This configuration will still provide all the normal controls, indications and interlocks for CRDM Cooler 1A-A to function. Because only 1 motor is able to run, CRDM Cooler 1A-A will only be operating at reduced capacity and will only be available for supplemental cooling mode. This TMOD will maintain CRDM Cooler 1A-A aligned in supplemental cooling mode, which is the required alignment for the Appendix R event, by administratively controlling 1-TCO-30-84 closed and 1-TC0-30-85 open. Its paired cooler, 1D-B, will be available for normal operation at full capacity to provide cooling to the CRDM shroud.

The normal controls, interlocks and indications for CRDM Cooler 1A-A will remain unchanged. The control circuit shown on 1-45W760-30-8 will be unaltered. 1-BKR-30-83/1 will operate as normal but will no longer be connected to any load because the primary disconnect finger clusters have been removed. All status light indications, interlocks, alarms, and manual and automatic control functions are unaltered and will continue to operate as designed. Annunciator

1-XA-55-5C-102A "CRDM COOLER FLOW LO" may alarm when CRDM Cooler 1A-A is operated in the bypass mode due to less than normal negative pressure at 1-FS-30-83A/B.

The required heat removal rate of the CRDM coolers only applies to cooling of the CRDM inside the Reactor Pit and does not account for any supplemental cooling from operating in Bypass mode. The required 2,600,000 BTU/hr of cooling is achieved by two CRDM Coolers operating together and aligned to the CRDM Shroud. This is unaffected by this TMOD. CRDM Cooler 1D-B plus either 1B-B or 1C-A may be aligned to the CRDM shroud to provide the required cooling during normal operation.

Calculations reference 15 and 16 specifies the equipment required for 10 CFR 50 Appendix R FSSD. Logic diagram K-37J-1 requires either 3 of 4 LCCs or 2 of 4 LCCs plus 2 of 4 CRDM Coolers in bypass mode to provide the required containment cooling to support FSSD. Calculation WBN-APS2-070 determines the temperature response of Containment following an Appendix R fire. Two base cases are analyzed in this calculation along with 4 sensitivity cases which consider off-normal airflow rates of the various containment coolers. The two base cases match logic diagram K-37J-1. Base Case 1 considers 2 LCCs plus 2 CRDM Coolers (at design flow rates). Base Case 2 considers 3 LCCs (at design flow rates) and no CRDM Coolers.

Sensitivity Cases 2, 3, and 4 consider potential CRDM Fan stage failures. Sensitivity Case 2 considers 2 LCCs plus 1 CRDM Cooler at design flow and 1 CRDM Cooler at 9,000 CFM. This was done to bound conditions when one of the two in-series CRDM Cooler fans fails and results in a potential operating point on the fan curve of approximately 9,000 CFM. Sensitivity Case 3 considers 2 LCCs plus 2 CRDM Cooler each at 9,000 CFM. Sensitivity Case 4 considers 2 LCCs plus a single CRDM Cooler at 9,000 CFM. The computed temperature response for each case is compared to its applicable Environmental Qualification (EQ) profile and results show that with three exceptions, all areas remain bounded by the EQ curve. The three exceptions are the lower reactor cavity, the upper reactor cavity, and the upper containment. A basis is provided in the analysis which justifies acceptability of the results.

This TMOD is bounded by the previous analysis performed in WBNAPS2-070. Therefore, implementation of this TMOD will be acceptable for Appendix R safe shutdown scenarios. Additionally, the Post Modification Testing (PMT) for this TMOD will operate CRDM Cooler 1A-A motor 2 in bypass mode, and verify the motor amps are within the expected range.

Because TMOD WBN-1-2017-067-001 is already installed which reduced the capacity of LCC 1D-B by 25%, the possibility of a reduced capacity LCC plus a reduced capacity CRDM Cooler must be considered. WBN-1-2017-067-001 contains analysis that show that even with a single LCC at 75% heat removal capacity and a single failed CRDM fan motor, the worst possible combination of containment coolers remain bounded by the original Sensitivity Case 4 of two LCCs plus only one CRDM cooler that results in a total cooling capability of 6,818,321 BTU/hr.

LCC 1D-B plus any one remaining LCC and CRDM Cooler 1A-A plus any one remaining CRDM cooler.

This case equates to approximately:

2,341,821 BTU/hr+ 3,122,427 BTU/hr+ 1,350,660 BTU/hr+ 573,467 BTU/hr= 7,388,375 Btu/hr of total cooling capability.

19. Evaluation: TMOD WBN-1-2017-244-001 R0, Evaluation R0

Activity Description:

During a walkdown of the Unit 1 Transformer Yard, it was discovered that the Main Bank Transformer (MBT) 1A High Voltage (HV) Bushing Lower Current Transformer (CT) had what appeared to be an insulating material or mastic on the outside of its enclosure. HV Bushing oil level was observed to be steady within the acceptable range and does not seem to be decreasing which indicates no loss of oil from the HV Bushing. The substance has been identified as insulating material, resin that is leaking out of one of the CTs.

The HV CT consists of a core and secondary coil assembled as a unit in an aluminum shield and completely embedded in a thermosetting casting resin. The manufacturing process leaves the resin surface open on what will become the bottom of the CT when it is installed. Mounting brackets are included as part of the shield, and extend slightly below its wall. This provides a space between the inner wall and the mounting surface when the current transformer is mounted on a flat cover.

On the MBTs, there are two CTs on the HV bushing. The Upper CT is mounted on top of the Lower CT. The open resin surface of the Upper CT is mounted against the top of the shield for the Lower CT. The material flow path on the outside of the Lower CT, discovered during the walkdown, indicates that the material came from the MBT 1A HV Upper CT and flowed down the outside of the Lower CT.

The HV Upper CTs from each MBT (1A, 1B, and 1C) provides input to the Schweitzer SEL-300G relays 1-RLY-244-S300/1 S1 and 1-RLY-244-S300/2S3. These relays are part of the Generator Backup Relay Protection Scheme and are arranged in a 2 out of 2 logic pattern to provide a 121 GB (Generator Backup distance relay) trip signal. The 121GB setting programmed into these relays will allow the relays to look from the HV Upper CT, through the Main Bank Transformers and to the Generator without encroaching on the Minimum Excitation Limiter (MEL) settings. This will allow the loss-of excitation protection to operate for most under-excitation conditions. Any faults which occur beyond the HV bushing CTs will be monitored by Transmission protection.

Relays 1-RLY-244-S300/1S1 and 1-RLY-244-S300/2S3 will obtain the current signal from the HV Upper CT as well as the voltage signal from the Bus Potential Transformer (PT) in the 500KV Yard and use this data to calculate the impedance of the corresponding circuit. A change in the voltage or current due to a fault will result in a change to the calculated impedance. The impedance value will indicate whether a fault has occurred between the HV Upper CT and generator. If the calculated impedance is less than the relay specified value, then this would indicate that the fault is between the HV Upper CT and the Generator and a trip signal will be generated from each relay. If the calculated impedance is greater than the relay specified value, then a trip signal will not be generated. The U1 turbine generator will still be protected by other relays for Generator Loss of Excitation and for Main Transformer Feeder Differential.

This TMOD will lift the turbine trip signal wiring from 1-RLY-244-S300/1S1 and 1-RLY-244-S300/2S3 which monitor the 121GB Relay function, thus preventing a spurious turbine trip signal from occurring due to the MBT 1A HV bushing Upper CT secondary windings coming into contact with the metal housing of the MBT 1A HV bushing Lower CT which would result in a reduction of the calculated circuit impedance by the relays which would produce a turbine trip signal.

The MBT 1A HV bushing Lower CT will remain in service and continue to monitor the current flowing through the MBT 1A HV bushing. This CT provides secondary current readings to the 187TF relay which monitors for phase imbalance between the A, B, and C phases of the Unit 1 MBTs, and will provide a turbine trip signal in the event of a detected phase imbalance.

Summary of Evaluation:

Performing this temporary modification lifts the turbine trip signal wiring from 1-RLY-244-S300/1S1 and 1-RLY-244-S300/2S3 which monitor the 121GB Relay function, thus preventing a spurious turbine trip signal from occurring due to the MBT 1A HV bushing Upper CT secondary windings coming into contact with the metal housing of the MBT 1A HV bushing Lower CT.

Lifting this turbine trip signal wiring from 1-RLY-244-S300/1S1 and 1-RLY-244-S300/2S3 has the potential of preventing a legitimate turbine trip signal in the event of a fault occurring between the Unit 1 MBT HV bushing CTs and the Unit 1 Generator. The prevention of the legitimate turbine trip signal from these two relays would also prevent a direct or immediate reactor trip as a result of this turbine trip signal. This could potentially result in a removal of the external electrical load from the Unit 1 turbine generator without a direct or immediate reactor trip.

If a legitimate fault between any of the Unit 1 MBT HV CTs and the Unit 1 Generator were to occur, thus removing the external load of the Unit 1 Generator, then other turbine generator protection, both mechanical and electrical (such as the Loss of Excitation monitored by the Beckwith Relays, and the Main Transformer 1 Feeder Differential Relays), would initiate turbine trip signals without the need for a turbine trip signal from either 1-RLY-244-S300/1S1 or 1-RLY-244-S300/2S3.

Because the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system, then lifting the turbine trip signal wiring from 1-RLY-244-S300/1S1 and 1-RLY-244-S300/2S3 to prevent a spurious trip signal will not result in a more than minimal increase in the consequences of an accident previously evaluated in the UFSAR.

Therefore it is acceptable to implement this temporary modification per plant procedures without obtaining a License Amendment.

20. Evaluation: TMOD WBN-2-2017-068-001 R0, Evaluation R0

Activity Description:

During Containment closeout walkdowns of U2RI, 2-SMV-68-548 was identified as leaking. This valve is a Kerotest Y-Type Globe Valve and the leak is coming from the Yoke/Body connection. 2-SMV-68-548 is the Hot Leg Loop #1 Sample Isolation Valve and is located directly off Hot Leg #1.

Based on a walkdown by the Boric Acid Program Engineer, the leak is expected to get worse over the course of the next cycle. Access to 2-SMV-68-548 is not available during Mode 1 or 2 because this valve is located inside the bio-shield (crane wall). Therefore, the valve will be isolated for Unit 2 Cycle 2. With the valve isolated and seated, the valve's diaphragm and yoke/body connection will not be subjected to RCS Pressure. Seating the valve will ensure the

yoke/body leak will not worsen over Unit 2 Cycle 2. This will remove the capability for Chemistry to sample the RCS from Hot Leg Loop #1. However, this TMOD does not impact the ability to sample the RCS from Hot Leg Loop #3, PZR Liquid Space, and the Inlet and Outlet of the Mixed Beds.

Summary of Evaluation:

The primary safety function of the sampling system is the containment isolation valves which close after a design basis accident. Containment isolation is unaffected by this TMOD.

The additional design functions of the impacted portions of the sampling system are to obtain and analyze samples of the RCS during normal operation as well as able to obtain and analyze samples of the RCS after a design basis accident. There are two sampling points in the RCS Hot Legs (one in Loop 1, and one in Loop 3) that are designed to operate after a design basis accident. There are two post accident sample points to assure that sampling can be accomplished in case one of the sample points fails. The TS require that the RCS be sampled every seven days to determine gross specific activity (detect failed fuel) and once every 14 days to perform 1-131 dose analysis.

The location where the RCS sample is required to be taken is not specified in the TS or the UFSAR and the procedure which meets the surveillance requirement allows for many locations to be used to sample the RCS (including both RCS Hot Legs, and points in the Chemical Volume and Control System and Residual Heat Removal System).

The temporary isolation of 1 potential RCS sample location does not impact the ability of the sampling system to meet its required Technical Specification or design function for both required RCS sampling and containment isolation.

21. Evaluation: TMOD WBN-1-2018-067-001 REV. 0, Evaluation R0

Activity Description:

One of the eight cooling coils on LCC 1D-B was found leaking at the beginning of U1R15. The leaking coil is the bottom reactor side coil, farthest from the room centerline (ladder) and is leaking at approximately 20 drops/min. LCC coils are Class C (ASME III, class 3) components and therefore this constitutes an ASME Code Class C component leek. Consequently, Unit 1 cannot enter Mode 4 (ascending) without repairing or isolating this leak (TR 3.4.5).

This evaluation covers the proposed modification to temporarily (for one operating cycle) install a blank-off plate to isolate ERCW flow from the leaking coil. This will result in seven of the eight coils of LCC 1D-B remaining fully functional. The blanking plates will be constructed of 0.1345" thick ASME III Class 3 stainless steel material to meet the ASME code requirements for this system and capable of withstanding the system design pressure and temperature of 160 psig and 130°F respectively. This change will not adversely impact air flow through any of the cooling coils nor will it impact continued ERCW flow to the remaining 7 coils associated with this cooler.

The cooling operation of the LCC cooling coils is not a safety related function; the coils' safety function is to maintain ERCW pressure boundary only. Operation of all four of the LCCs is needed during the summertime/fall time period when ERCW temperatures are at their highest

during the year. Operation of these coolers is needed to maintain containment air temperature of 120 °F during normal plant operation.

Summary of Evaluation:

Proper operation of the LCC cooling coils is not a safety related function and not required for mitigation of any UFSAR Chapter 6 or 15 accidents. The blanking plates will be designed to meet the design conditions of the ERCW system. The design basis functions (movement of air in lower containment) of the LCC 1D-B is maintained and not adversely impacted by this proposed design change. Therefore this change does not result in any new accidents or malfunctions, and does not result in increased frequency or consequences of accidents or malfunctions evaluated in the UFSAR. In addition, no fission product barriers are challenged by this change.

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.

22. Evaluation: TMOD WBN-1-2018-067-002 REV. 0, Evaluation R0

Activity Description:

Three of the eight cooling coils on Lower Compartment Cooler (LCC) 1A-A were found leaking during U1R15. The bottom coil, on the reactor-facing side, closest to the room access ladder was found leaking as well as the top and bottom coil on the side farthest from the reactor facing side, farthest from the room access ladder. LCC coils are Class C (ASME III, class 3) components and therefore this constitutes an ASME Code Class C component leak. Consequently, Unit 1 cannot enter Mode 4 (ascending) without repairing or isolating this leak (TR 3.4.5).

This evaluation covers the proposed modification to temporarily (for one operating cycle) install blank-off plates to isolate ERCW flow from the leaking coils. This will result in five of the eight coils of LCC 1A-A remaining fully functional. The blanking plates will be constructed of 0.1345" thick ASME III Class 2 stainless steel material to meet the ASME code requirements for this system and capable of withstanding the system design pressure and temperature of 160 psig and 130°F respectively. This change will not adversely impact air flow through any of the cooling coils nor will it impact continued ERCW flow to the remaining 5 coils associated with this cooler.

The cooling operation of the LCC cooling coils is not a safety related function; the coils' safety function is to maintain ERCW pressure boundary only. Operation of all four of the LCCs is needed during the summertime/fall time period when ERCW temperatures are at their highest during the year. Operation of these coolers is needed to maintain containment air temperature of 120 °F during normal plant operation. Due to WBN-1-2018-067-001, this evaluation considers any cumulative impact caused by a single coil being out of service on LCC 1D-8.

Summary of Evaluation:

Proper operation of the LCC cooling coils is not a safety related function and not required for mitigation of any UFSAR Chapter 6 or 15 accidents. The blanking plates will be designed to meet the design conditions of the ERCW system. The design basis functions (movement of air in lower containment) of the LCC 1A-A is maintained and not adversely impacted by this

proposed design change. Therefore this change does not result in any new accidents or malfunctions, and does not result in increased frequency or consequences of accidents or malfunctions evaluated in the UFSAR. In addition, no fission product barriers are challenged by this change.

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.

23. Evaluation: CR 1356846, Evaluation R0

Activity Description:

The proposed change is an Accept-As-Is evaluation for CR 1356846. The condition being evaluated is the leaving an unknown pliable substance attached to the bottom nozzles of 29 fuel assemblies when Unit 2 starts up from U2R1.

CR 1356846 documents that during core offload, debris was found on the bottom nozzle of multiple fuel assemblies. 29 of these fuel assemblies are scheduled to be reused in the core for U2C2. The following fuel assemblies are to be reused:
M05, M32, N13, N43, M11, M35, N14, N51, M12, M36X, N20, N59, M14, M40, N25, N61, M16. N29, N63, M49, N35, N64, M24, M53, N38, M29, M55, M21, and N41.

The foreign material was observed in the flow holes of the bottom nozzles of about 40 fuel assemblies during the core off-load following Cycle 1 at Unit 2. There are no reports of the foreign material at any other location on the fuel assemblies. Multiple attempts were made to retrieve a sample of the debris for analysis without success. The unknown pliable material is stuck to the fuel assemblies with one or two flow holes per assembly potentially blocked.

Based on the Westinghouse evaluation performed for this issue no adverse impacts to the reactor vessel internal structure, the vessel itself, or other components in the RCS system are anticipated either short term or in the long term.

Summary of Evaluation:

The Westinghouse evaluation of the effect of the foreign material on the bottom nozzle of the fuel rod assemblies or in the RCS during operation determined that, although there are ways in which the foreign material could induce detrimental effects, the likelihood of an accident or malfunction is not increased as such magnitude of detrimental impact is not plausible. The Westinghouse evaluation expects no more-than-minimal increase in the consequences of any accident or malfunction because both cases are bound by the pre-existing assumptions for failed fuel (i.e., all of the rods in the assembly are assumed damaged) and for the highest-worth Rod Cluster Control Assembly (RCCA) being completely retracted. The pieces of foreign material are too small and too pliable to induce, or cause other debris that could induce, RCS inventory loss or component damage such that a new type of accident or malfunction is created

24. Evaluation: CR 1357401, Evaluation R0

Activity Description:

The proposed change is an Accept-As- Is evaluation for CR 1357401. The condition being evaluated is as follows:

During U2R1 the nylon tip (nose cone) of an eddy current bobbin probe broke off in Steam Generator 4 and could not be retrieved. This lost nose cone is from ZETEC®1 Part No 10025205 and is estimated to be approximately one inch in length with a weight of 10 to 25 grams.

Westinghouse has performed assessments supporting operation with primary side loose parts comprised of nylon for several Westinghouse-designed plants. Other nylon loose parts previously evaluated for other plants include eddy current probe feet, tie wraps, and bushings from within a stainless steel nut. Because nylon softens and disperses into the reactor coolant at operating temperatures, there are no significant concerns for operability with nylon objects in the primary side.

Based on the Westinghouse evaluation performed for this issue no adverse impacts to the reactor vessel internal structure, the vessel itself, or other components In the RCS system are anticipated either short term or in the long term.

Summary of Evaluation:

The Westinghouse evaluation of the effect of the nylon probe tip or fragments of the probe tip that may be generated as they migrate through the RCS will have no adverse effects on systems, components, or their design functions. Contaminants from the melting of the nylon material at operating temperatures will be dispersed in the RCS primary coolant and have an insignificant effect on the RCS water chemistry.

Therefore, there are no operational issues associated with the nylon material from the probe tip in the primary side of the RCS.

25. Evaluation: PNNL-TTP-7-613 Rev. 2, Evaluation R0

Activity Description:

Pacific Northwest National Laboratories (PNNL) informed TVA of changes to the currently approved Mark 9.2 design Tritium Producing Burnable Absorber Rod (TPBAR) assembly. The changes will be implemented in two Lead Use Assemblies (LUAs) inserted into WBN 1 Cycle 16. The changes made to the TPBAR LUAs are as Follows:

Rod C16-1 uses a 3-inch stack of 0.025-inch wall thickness pellets with all other components being of Mark 9.2 design. The 0.025-inch wall thickness is a thinner pellet wall thickness than the Mark 9.2 design. The 3-inch thinner wall pellet stack will be located near peak fluence resulting in a higher burnup with a limit not to exceed 21.2% total lithium. The pellet OD will remain the same as a standard pellet of 0.302-inch. The remainder of pellets in the 132-inch absorber stack will be standard pellets.

Rod C16-2 uses a full length thicker getter, with a full column of 0.030-inch thick wall pellets, with all other components a Mark 9.2 design. The finished plated getter wall dimensions changed from OD <or equal to 0.328" and ID> or equal to 0.307", to OD <or equal to 0.328" and ID> or equal to 0.303". This change in getter wall dimension will accommodate an increase in Zirconium (Zr) mass from a nominal 0.703 gm/inch plus or minus 0.198 gm/inch to a minimum Zr mass greater than or equal to 0.814 gm/inch. The nickel platting on the getter remains the same as the Mark 9.2 design. The pellet wall thickness of 0.030" is thinner than the Mark 9.2 design and will be used for all pellets in the 132-inch absorber stack. Burnup will remain within

the current design limits. The pellet OD was reduced from 0.302" to 0.298" to accommodate the thicker getter in the same diameter TPBAR. In addition, a minimum getter wall thickness is identified greater than 0.0081-inches that was not previously specified for Mark 9.2 TPBARs.

The LUA TPBARs do not change the TPBAR design function for reactivity control and tritium production. The purpose of the two LUA TPBARs is to investigate refinements, of 0.025 inch and 0.030 inch wall pellets and a thicker (>0.0081-inch wall thickness) getter, that provide design enhancements that may improve margin to design criteria and allow higher tritium production capabilities, while continuing to meet tritium retention performance. The design changes do not represent new features outside of the Approved Mark 9.2 TPBAR Design, they are refinements. The changes are consistent with the conclusions reached by the NRC in the Safety Evaluation Reports (SERs) for TVA's License Amendments No. 40, 48, 67, 77, and 107.

Because the potential exists for the design changes included in LU Rod C16-1 and C16-2 to adversely impact the expulsion of pellet material during a LBLOCA, LUA rod C16-1 will exceed the pellet burnup limit for a 3" stack of pellets and the maximum zirconium mass of the getter tube for LUA rod C16-2 will increase from a value of 0.90 gm/in to 1.30 gm/in., a 50.59 evaluation is recommended.

Summary of Evaluation:

The two LUA rods, C16-1 and C16-2, are within the current design limits for tritium production. The pellet burnup limits remain within TPBAR design limits. The tritium releases and doses that could result from their irradiation are bounded by current analyses. Further, the LBLOCA burst performance of these two LU rods will not impact the core's response to a LBLOCA if all the pellet material (negative reactivity) is ejected, thus no License Amendment is required.

26. Evaluation: SAR Change Package 02-010, Evaluation R0

Activity Description:

The uncontrolled boron dilution event described in UFSAR Section 15.2.4 was reanalyzed to determine the minimum ratio of the initial (Cb_1) to critical (Cb_c) boron concentrations for each mode of operation that would ensure 15 minutes would be available between the alarm for an uncontrolled boron dilution event and the time of complete loss of shutdown margin. Previous evaluations using a more restrictive ratio calculated more than 15 minutes of time available for operator action.

Summary of Evaluation:

By meeting the acceptance criterion as defined in the Standard Review Plan and SER, it was concluded that there is sufficient time available to terminate the dilution prior to loss of shutdown margin. Therefore the event does not result in RCS overpressurization, fuel rod damage, or more serious accident.

27. Evaluation: WBN Unit 2 Cycle 2 COLR Rev. 0, Evaluation R0

Activity Description:

The proposed activity replaces the Unit 2 cycle 1 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 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 Unit 2, starting with the Cycle 2 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 method used for Cycle 1. It is shown that the change does not constitute departure from a method of evaluation; therefore, 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.

10 CFR 50.59 Summaries not Previously Submitted

The following 10 CFR 50.59 summaries were identified that should have been previously submitted to the NRC. This issue has been captured in the site Corrective Action Program.

28. Evaluation: DCN 57700 Rev. A, Evaluation R0

Activity Description:

DCN 57700 issues the 0-47E235 drawing series to document environmental parameters for two unit operation. Included are changes to some areas which contain Unit 1 operating equipment required for 10 CFR 50.49 qualification. The values (temperature, pressure humidity, etc.) documented on these drawings are from calculations WBNAPS4004 and WBNAPS4008. Changes in values (temperature, pressure humidity, etc.) makes it necessary to revise the associated EQ Binders to show the acceptability of the Unit 1 equipment once Unit 2 begins operation.

Summary of Evaluation:

DCN 57700 is issuing revised drawings depicting the normal and post accident environmental conditions. Issuing of these drawings does not result in any 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, RCS 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 drawing and calculation revisions; and there is not a change in evaluation methodology.

The conclusion is that the change can be implemented per existing processes without obtaining a licensing amendment.

29. Evaluation: ECP 58314 Rev. A, Evaluation R0

Activity Description:

Background:

The existing analog TDAFW Governor Control System is being upgraded to a digital system in response to industry obsolescent issues associated with the end of life and lack of continuing vendor support of the existing Woodward EG-type turbine controller. Because of the widespread use of digital technology in the non-nuclear industry, a suitable analog replacement for the existing system is not available. This modification will replace the TDAFW turbine controls with a new, industry standard digital control system similar to that being used on reactor feed pump and auxiliary feed pump control systems in the nuclear industry.

Description:

The existing Unit 1 TDAFW speed governor system will be replaced in its entirety by this modification. The replacement governor system will replicate the functionality of the existing speed control and electrical overspeed trip components of the existing system, as well as incorporating the flow control function of the existing Foxboro Spec 200 components located in panel 1-L-381A. Incorporating the flow controls functions within the new governor system will simplify the control system and reduce the number of active components required to perform the TDAFW design function, as well as allowing the use of a standard replacement governor design across the TVA nuclear fleet.

This modification will remove and replace the existing governor panel 1-L-326 in its entirety, and install a new actuator positioner panel 1-L-326A to house the actuator positioner (controller) and associated components. The new 1-L-326 panel will be installed in the same location as the existing panel. The new 1-L- 326A panel will be installed near the new 1-L-326 on the outside wall of the TDAFW pump room.

The existing two magnetic speed pickups 1-SE-46-56 and 1-SE-46-57 on the AFW turbine will be replaced with similar magnetic pickups and vendor supplied cables to the TDAFW skid mounted junction box. Existing speed pickup wiring from the TDAFW skid mounted TB3 to panel 1-L-326 will be re-used.

Existing TDAFW MCR speed indicator 1-SI-46-56A and speed setpoint indicator 1-XI-46-54A on panel 1-M-4 will be replaced with new indicators to accept a 4-20mA signal from the replacement system. The indicated parameters, scaling, and labeling are unchanged.

Existing TDAFW MCR pump demand indicator 1-XI-46-57 on panel 1-M-4 will be replaced with a new TDAFW flow setpoint indicator that will accept a 4-20mA signal from the replacement system. This indicator will now display the governor flow setpoint, and will be labeled to reflect the revised displayed parameter. The new 1-XI-46-57 panel meter scale will be similar to existing flow indicator 1-FI-3-142A. The existing 10-50mA/10-50mA 1E/non-1E signal isolator 1-XM-46-57 located in panel 1-L-381 will be replaced with a similar 4-20mA/4-20mA 1E/non-1E signal isolator to drive this indicator.

Existing local governor demand signal isolator 1-FM-46-57 located in panel 1-L-381 is deleted by this modification as its function is no longer required with the revised governor controls.

Existing TDAFW MCR flow indicator 1-FI-3-142A on panel 1-M-4 and ACR flow indicator 1-FI-3-142C on panel 1-L-10 are unaffected by this modification.

A new ICS data acquisition device 1-PLC-261-5465, as well as an associated 24VDC power supply 1-PX-261-5465, and fuses 1-FU2-261-5465/1 and 1-FU2-261-5465/2 will be installed in existing panel 0-JB-292-5465 located at coordinates A8-T on elevation 692 of the Auxiliary building. This new data acquisition device will be connected to existing ICS network switch 0-XS-261-5042 located in panel 0-JB-292-5465.

New class 1E signal isolators will be installed in panel 1-L-381 to provide for 1E/non-1E isolation of analog signals from the replacement system to the ICS. These signals will be connected to the new ICS data acquisition device 1-PLC-261-5465 in panel 0-JB-292-5465.

Spare relay contacts will be utilized to provide digital status signals from the replacement system to the integrated computer system. Coil-to-contact isolation will be used to provide non-1E status signals. These signals will also be connected to the new ICS data acquisition device 1-PLC-261-5465.

The new TDAFW control system is provided with local manual control capability at the TDAFW turbine with no electrical power (AC or DC) available by manual operation of the new governor valve actuator jacking screw. Although available, this "Black Start" capability is not credited for any design basis events. The governor valve (1-FCV-1-52) can be manually positioned to control AFW speed with the actuator jacking screw. The existing "Black Start" methodology of leaving 1-FCV-1-52 in the normal full open state, and controlling AFW turbine speed by manually positioning the trip and throttle valve 1-FCV-1-51 can still be utilized.

The new system is powered from the existing 125V DC sources the same as the existing control system. This modification will result in an increase to the 125V DC battery loading for batteries III (normal feed) and IV (alternate feed).

A trip and throttle valve (1-FCV-1-51) position signal is provided to the governor controller via 1-ZS-46-57C (LS-6) limit switch. As the 1-FCV-1-51 valve unseats (not full closed) and engages the limit switch, the contact is made up and initiates the TDAFW governor start sequence. The output of this circuit will initiate a controlled startup of the governor control valve (1-FCV-1-52); thus minimizing the potential for a possible electrical or mechanical overspeed trip. This limit switch signal, in parallel with a >100 rpm speed contact from the 505 governor, will be used to enable the new actuator positioner. Enabling the actuator positioner will stroke the governor valve from its normally open/fail state of full open to the position demanded by the governor.

New cabling and raceway will be installed from the replacement 1-L-326 governor panel to adjacent panel 1-L-381, and from panel 1-L-381 to panel 1-L-381A. New cabling and raceway will be installed from the replacement 1-L-326 governor panel to the new 1-L-326A positioner panel. New cabling and raceway will be installed from the new 1-L-326A positioner panel to the new governor valve actuator on the TDAFW skid. New cabling and raceway will be installed from the 1-L-381 panel to ICS panel 0-JB-292-5465 located at A8-T on elevation 692' of the Auxiliary Building.

Combining multiple functions into a single digital device:

Various individual analog components are being replaced with a single digital device that combines the existing functions into a single digital controller. The new digital controller increases the complexity of the governor and creates the potential for different failure modes.

Summary of Evaluation:

This evaluation has determined that the TDAFW system will continue to meet its design and licensing bases requirements following the implementation of the proposed modification that converts to a digital governor control method for the system.

Because the new TDAFW System components are more reliable than the existing components and no new system level failure mode effects are introduced, the proposed modification does not result in more than a minimum increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The new equipment being installed will not initiate any new system malfunctions. Credit is taken for the TDAFW System for the successful mitigation of the following transients, special events and accidents. The AFW system supplies, in the event of a loss of the MFW supply, sufficient feedwater to the SGs to remove primary system stored and residual core energy. It may also be required in some other circumstances such as the evacuation of the MCR, cooldown after a LOCA for a small break, maintaining a water head in the SGs following a LOCA, a flood above plant grade, ATWS event, and 10 CFR 50, Appendix R, Fires. The TDAFW System will not adversely impact any of the systems that have a dynamic interface with the TDAFW System. Namely: Condensate Storage Tanks; ERCW; Main Steam; Feedwater; and 125 Volt DC Power Systems. Therefore, the modification does not result in more than a minimum increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Performance requirements associated with core cooling are unaltered such that fuel integrity will be maintained and the UFSAR analysis of radiological consequences remains bounding. The TDAFW Systems ability to mitigate any postulated design basis accidents will not be decreased. The new equipment will not initiate any new accidents. The modification will not impair or prevent the ECCS from mitigating the consequences of any design basis accidents. Therefore, this activity does not result in more than a minimum increase in the consequence of an accident previously evaluated in the UFSAR.

Failure or malfunction of the new equipment will not prevent or affect the ability of safety related systems or systems important to safety to respond to the accidents describe in the UFSAR. Therefore, implementation of the proposed modification does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR will not be increased. The potential malfunctions of the modified equipment are bounded at a system level in the UFSAR. Therefore, the possibility for an unanalyzed malfunction of an SSC important to safety or an accident of a different type than any previously evaluated in the UFSAR are not created.

As described in the UFSAR accident analysis, no malfunction of the AFW System can cause a transient sufficient to damage the fuel barrier or exceed the nuclear limits as required by the safety design basis. As described in the UFSAR FMEA Tables 10.4-3 and 10.4-4 no new failure modes are created by replacement of TDAFW governor valve controller. The proposed modification does not adversely impact the technical attributes supporting this conclusion.

Therefore, the possibility for an unanalyzed malfunction of an SSC important to safety or an accident of a different type than any previously evaluated in the UFSAR are not created.

The new digital equipment does not necessitate a revision or replacement of any currently used evaluation methodology for the TDAFW System. The modification does not result in a departure from the method of evaluation described in the UFSAR in establishing the design bases or in the safety analyses.

Guidance for evaluation of digital upgrades is contained in NEI 01.01/EPRI TR-102348, Guideline on Licensing Digital Upgrades, March 2002. {NPG Electrical Engineering Design Guide DG-E18.1.25, Digital System Development, Procurement and Implementation require that the supplemental questions from NEI 01-01 be addressed within a conceptual design.} NEI discusses the use of an FMEA. In line with the graded approach referenced in NEI 01.01, an FMEA was written to identify those malfunctions important to safety the digital controls could cause.

NRC IN 2010-10 discusses a digital modification to the LaSalle non-safety-related control rod drive system. The NRC noted that LaSalle's application of 10 CFR 50.59 did not answer all the questions in Appendix A of NEI 01.01 in the associated 50.59 Evaluation. In the Criteria Evaluation below, the questions in Appendix A of NEI 01.01 are provided in italics and answered for the proposed digital upgrade. Following the NEI 01.01 questions the criterion question is answered for the analyzed activity.

This evaluation concludes that implementation of the modification does not require a Technical Specification change, does not require a License Amendment, and therefore may proceed without NRC approval.

30. Evaluation: DCN 60060 Rev. A and DCN 61422 Rev. A, Evaluation R0

Activity Description:

The NRC issued Order EA-12-049 in response to the events at the Fukushima Daiichi nuclear plant in Japan. The Order was concerned with nuclear plant response to an extended loss of AC power event (ELAP) and loss of normal access to the ultimate heat sink (UHS) due to beyond design basis external events (BDBEE). The NEI proposed strategies to mitigate these events in NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide, which was endorsed by the NRC.

DCN 60060, Installation of AFW Supply Tank (AFWST), was developed in response to NRC Order EA-12-049 and the interpretation of the order per NEI 12-06. Specifically, NEI 12-06 Section 3.2.2(5) states that the condensate storage tanks (CSTs), if available, should be used as the first alternative to normal cooling and makeup inventories. The existing Unit 1 and Unit 2 CSTs were designed to the AWWA D100 code and are unanchored. Review of the seismic capability of the existing Unit 1 and Unit 2 CSTs by the original tank supplier determined that the tanks could not be seismically qualified and resulted in the recommendation by the supplier that the existing tanks be replaced. Instead, TVA chose to install a new dedicated tank to store a sufficient volume of condensate quality AFW for two units during BDBEE as described in NRC Order EA-12-049. The existing CSTs will remain as installed and will continue to be used for normal operation, and the primary source of water for the AFW pumps, as described in the UFSAR.

Enclosure

The purpose of this modification (hereafter referred to as DCN 60060) is to install a new seismically designed 500,000 gallon AFWST, including supply piping and isolation valves which connect to the existing Unit 1 and Unit 2 Condensate System supply piping. This tank and associated piping will supply the Unit 1 and Unit 2 AFW System pumps. The new dedicated AFWST will store a sufficient volume of condensate quality AFW for two units during a BDBEE (as described in NRC Order EA-12-049 and NEI 12-06). The addition of the tank and connecting piping will be accomplished using two DCNs. The first, DCN 61422, will install a tie-in connection to the Unit 1 AFW supply piping so that this DCN, DCN 60060, can be installed and connected to the Unit 1 piping without loss of operability.

The scope for DCN 60060 includes design and installation of:

a) AFWST foundation to interface with DCN 62324 (Micropile deep foundation)
b) 500,000 gallon AFW Supply Tank with stair access to tank roof, manways, drain tap, sampling tap, FLEX connections (as defined in NEI 12-06), conservation vent, and overflow.
c) Nitrogen sparger in the AFWST with piping connected to a tie-in point, provided by DCN 59234, on the Nitrogen portion of the Waste Gas Disposal system.
d) Common supply piping from the AFWST, connecting to the Unit 2 condensate line supplying the Unit 2 AFW Pumps and to the Unit 1 tie-in piping installed by DCN 61422.
e) AFWST level instrumentation with output to Distributed Control System and Integrated Computer System (DCS/ICS), with the DCS/ICS connection per DCN-56905 by Sargent and Lundy. There is an additional AFWST mechanical pressure/level indicator located in the missile protected valve vault. This instrument provides the backup tank level information in case of loss of power to the level transmitter.
f) Air Operated Valve (AOV) isolation valve with associated limit switches, air supply, solenoid valves, and a check valve at the Unit 2 tie-in to existing condensate piping, similar to that provided for Unit 1 in DCN 61422.
g) AFWST fill line connection from the condensate transfer pump discharge piping.
h) Grounding, lighting, and cabling as required.
i) Pressure taps, root valves and switches on the Unit 2 condensate piping from the Unit 2 CST (note that the U1 pressure tap and root valve are installed per DCN 61422). ,
j) Flushing, grouting, plugging, and abandonment of the Fuel Oil Drain and Lube Oil Drain piping in the vicinity of, and underneath, the AFWST foundation.

There is an UFSAR change package associated with this activity. The package describes that" ... the 500,000 gallon AFW supply tank serves as an additional condensate-grade source of water. The AFWST is normally isolated from the condensate piping, which provides water to the AFW Pump suctions, by an AOV. The AOV will open upon a low pressure signal from the upstream condensate piping, a loss of AC power, or a loss of control air. This sequence is not an Engineered Safety Feature and is not credited in the Safety Analyses." This DCN supports the TVA action plan for activities related to the Fukushima event.

Because this DCN 60060 interfaces with DCN 61422, the scope of this review also discusses the impact to the plant with both DCNs implemented.

The purpose of DCN 61422 is to install a new connection, check valve, and isolation valve to the Unit 1 AFWP supply line so that the second DCN, DCN 60060, can be installed and connected to the Unit 1 piping without incurring inoperability of the condensate supply.

The scope for DCN 61422 is limited to the addition of a check valve and tee downstream of existing isolation valve 1-ISV-002-0504, and an air-operated isolation valve connected to the

new branch line from the tee. DCN 60060 will connect the AFWST to this branch line. Additionally, DCN 61422 will add a 1/2 inch instrument isolation valve to the existing 12" condensate piping.

The check valve and branch connection tee installed by DCN 61422 will be installed on the existing 12 inch Unit 1 Condensate System supply piping, between isolation valves 1-ISV-002-0504 and 1-ISV-003-0800, in the piping from the Unit 1 CST (1-TANK-002-0229) to the Unit 1 AFW Pumps. The tie in will be located in the open bay of the Control Building, floor elevation 708'. The piping and other components for the tie· in will be Class H non-quality related.

Design Basis Accidents Involved

This modification provides an alternate supply of condensate grade water to the AFW) System through the Condensate System piping. In doing so, all accidents relative to the Condensate System are also applicable to the AFWST and associated piping.

From TS Bases B3.7.6-Condensate Storage Tank:
"The CST provides the preferred cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). However, the ERCW System provides the safety grade water source to meet a DBA should the CST become unavailable."

From UFSAR Section 10.4.9.1 -
"The auxiliary feedwater (AFW) system supplies, in the event of a loss of the main feedwater supply, sufficient feedwater to the steam generators to remove primary system stored and residual core energy. It may also be required in some other circumstances such as the evacuation of the MCR, cooldown after a LOCA for a small break, maintaining a water head in the steam generators following a LOCA, a flood above plant grade, Anticipated Transient Without Scram (ATWS) event, and 10 CFR 50, Appendix R, Fires."

Credible Failure Modes

The new AFWST and associated piping and valves have been designed to be "seismically robust," however, they are not qualified to withstand or mitigate any chapter 15 accident. The CSTs, which are the normal supply of cooling water to remove decay heat and to provide cooling water to the steam generators following all events in the accident analysis as discussed in the UFSAR, Chapter 15, are also not qualified to withstand or mitigate any chapter 15 event. The current failure modes and effects analysis for AFW Motor Driven and Turbine Driven Pumps assumes that the tank discharge is plugged. The effect is none, in that on loss of the condensate supply, the essential raw cooling water system supply is automatically provided.

Summary of Evaluation:
This modification is an addition to the AFW system, utilized to mitigate chapter 15 accidents. The volume of water supporting TS limits is unchanged, and additional water will be made available by the AFWST.

This modification is an addition to the AFW system, utilized to mitigate chapter 15 accidents. The AFW system is analyzed for a Moderate Energy Line Break (MELB), and this addition does not increase the frequency of a MELB in this system, nor does it increase the impact. This modification does not result in any increase to the frequency of occurrence of any accident

previously evaluated in the UFSAR. The systems associated with this modification cannot initiate any Chapter 15 event besides a MELB.

The 50.59 screening review, performed for this modification, concluded that the proposed modification was not adverse to the design function of the CST and associated piping, as described in the UFSAR and TS, except for the addition of a new check valve, installed in the CST discharge piping. The probability of failure of this check valve has been minimized by designing the installation to comply with Electric Power Research Institute (EPRI) standards, and by entering the valve into the WBN Augmented Inservice Testing Program. A PRA evaluation was performed and concluded that there is no significant impact to Core Damage Frequency (CDF) or Large Early Release Frequency (LERF) given the failure of the check valves. The failure of the non-safety and non-seismic CST's is already assumed and described in the UFSAR table 10.4-4 and 10.4-5, Failure Modes and Effects Analysis, TD Pump Subsystem and Motor Driven (MD) Pump Subsystem. The table describes that "on a loss of condensate supply, the ERCW system supply is automatically provided." The results of the FMEA, as presented in the UFSAR still remain bounding for the system, regardless of the failure of the check valve to open.

The CST provides the preferred cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the UFSAR, Chapters 6 and 15. However, the ERCW System provides the safety grade water source to meet a DBA should the CST become unavailable.

The addition of the check valves, tee, AOVs, and AFWST does not affect the settings of the pressure switches used to transfer from condensate to ERCW, and the switchover from condensate to ERCW is not hindered by this modification. This modification will only benefit the plant by providing additional condensate grade water, available for use prior to ERCW switchover.

The original NRC approval of this system states that the Condensate system was evaluated and found to have no functions necessary for achieving safe reactor shutdown or for accident prevention or mitigation.

In conclusion, the scope included in DCNs 60060 and 61422 does not require prior NRC approval.

31. Evaluation: ECP 65345 Rev. A, Evaluation R0

Activity Description:

The proposed change is to (a) revise the Offsite Dose Calculation Manual (ODCM) Table 1.1-2 to delete Item 3d and Applicability Notation 6 regarding isokinetic flow control operability and the associated requirement to manually restore isokinetic flow for the shield building ventilation radiation monitors
1-RE-90-402 and 2-RE-90-402, and (b) revise impacted plant procedures. There are no physical changes to the Shield Building Ventilation System and its radiation monitors 1-RE-90-402 and 2-RE-90-402.

Summary of Evaluation:

The proposed changes are procedural changes only. Based on the conclusions of calculation

MDQ0000902014000494R0, there is no impact on the sample accuracy of the radiation monitors 1-RE-90-402 and 2-RE-90-402 and no change is required to the transmission factors during isokinetic sample flow conditions. The concentrations in the Shield Building Ventilation System process flow stream and in the sample probes are nearly identical during an isokinetic sample flow conditions. Thus, the deletion of Item 3d, Isokinetic Flow Control Equipment and Applicability Notation 6 regarding isokinetic flow control operability and the requirement to restore isokinetic flow manually will have no adverse impact on detection of releases from the Shield Building Ventilation System to the environment or the capability to meet the regulatory limits specified in WBN TS, Section 5.7.2.3.a.2 for normal operation or during accident conditions. There is no physical changes to the Shield Building Ventilation System or the radiation monitors (1 -RE-90-402 and 2-RE-90-402) or a change to the radiation monitor transmission factors. Therefore, there will be no impact on the function of the radiation monitors or their capability to reliably monitor and measure the effluents from the Shield Building Ventilation system within regulatory limits prescribed in WBN TS, Section 5.7.2.3.a.2 and 10 CFR 100, 10 CFR 50.67 (Fuel Handling Accident only) and GDC 19 for design basis accidents.

32. Evaluation: DCN 66145 Rev. B, Evaluation R0

Activity Description:

Unit 1 has periodically experienced cracking of the last stage (L-0) blades of the Low Pressure "C" (LP-C) Turbine section of the main turbogenerator. WBN is replacing the LP-C Turbine with a Siemens advanced disc rotor design. The proposed rotor design would be optimized for 13.9m² L-0 damped blades (13.9 m² exit area downstream of the L-0 blades).

A Blade Vibration Monitor (BVM) system, also supplied by Siemens, was previously installed under TMOD WBN-1-2015-047-001 to support vibration monitoring of the current LP-C Turbine. The BVM system cabinet (with enclosed hardware) will be permanently mounted and the required conduits and wiring will be routed as necessary to connect to the new LP-C Turbine BVM probes. Conduit and wiring will also be installed to provide a permanent power supply (Lighting Panel LC106, circuit breaker 5) for the BVM.

Based on the LP-C Turbine modification described above, the ICS will be programmed to provide a new backpressure alarm curve. Also, loop alarm unit, 1-PS-2-10 setpoint will change from 5.3"HgA incr. to 6.6"HgA incr. The alarm unit also actuates annunciator window no. 3A-46C.

Summary of Evaluation:

The method of evaluating turbine missiles in UFSAR 3.5.1.3 is being revised to incorporate guidance given by the NRC in NUREG-0800. The current UFSAR evaluation is based on methodology calculating strike and damage probability (P_2xP_3) along with P_1 (probability of turbine failure resulting in the ejection of turbine rotor or internal structure fragments through the turbine casing). Because of uncertainties associated with calculating P_2 and P_3 as documented in NUREG-0800, the NRC no longer encourages applicants to calculate P_2 and P_3 or their product, but now accepts a product of strike and damage probabilities of 10^{-3} per year per plant for a favorably oriented turbine (like WBN).

Siemens Missile Probability Analysis Report CT-27547 Revision 1 confirms that the low pressure turbine with the current LP-A and LP-B Westinghouse rotors and the upgraded Siemens 13.9m^2 damped element LP-C rotor complies with the NRC P_1 limit of 10^{-4} per year for

a maximum of 12 years. Combining the Siemens determined P_1 with the NRC approved value of 10^{-3} for strike and damage probability for a favorably oriented turbine, provides justification that the new configuration will meet the NRC required P_4 limit of 10^{-7} per year.

Based on the Standard Review Plan (SRP) requirements for turbine missiles and the Siemens Corporation Missile Probability Analysis Report, this change is (a) based on sound engineering practice, (b) appropriate for the intended application, and (c) within the limitations of the applicable NRC requirements (NUREG-0800 and associated reference documents). Accordingly, this modification may be implemented without prior NRC approval.

33. Evaluation: PDO for PER 946856 Rev. 0, Evaluation R0

Activity Description:

PER 946856 was initiated on October 17, 2014 and identified the condition where the ERCW temperature control valve 0-TCV-067-1051-A is not modulating to control the flow of cooling water from the ERCW system to the condenser for the MCR Chiller A (0-CHR-031-0080-A). On October 17, 2014, WBN Operations entered TS LCO 3.7.11, Condition A, One Control Room Emergency Air Temperature Control System (CREATCS) train inoperable upon discovery of the Temperature Control valve (TCV) failed in the open position. The failed open TCV results in increased cooling of the chiller refrigerant due to the increased flow of cooling water to the condenser. The MCR Chiller A is currently operating outside the normal inservlce operating ranges for discharge pressure and oil temperature. The normal operating range for the oil temperature is 146 -150°F. The normal condenser refrigerant suction pressure is 60 - 70 psig and the refrigerant discharge pressure is 200 - 225 psig. The documented values for the chiller on October 17, 2014, at approximately 1330 hours were 120°F oil temperature, 140 psig condenser refrigerant discharge pressure, and 61 psig condenser refrigerant suction pressure.

The MCR Chiller A is itself performing its design basis functions; the problem involves the failure of the TCV to control flow to the cooler.

The safety-related function of 0-TCV-067-1051-A is to regulate ERCW flow through the condenser of the Train A MCR Chiller A. The ERCW flow regulation provides controlled cooling of the chiller's condenser which supports reliable chiller operation.

The Train A ERCW header supplies cooling water flow to MCR Chiller A. The safety-related functions of temperature control valve and associated MCR Chiller A are to: 1. maintain acceptable temperatures for protection and reliable operation of plant controls and equipment; 2. provide for safe, uninterrupted occupancy of the Main Control Room Habitability Zone (MCRHZ) during normal, accident, and post-accident recovery conditions. In addition, MCR Chiller A has a non-safety related function to maintain the Control Building spaces (except spreading room) at the design relative humidity of approximately 50%. 0-TCV-067-1051-B and MCR Chiller A are part of the MCR Air-Conditioning System, which Is included In the Unit 1 TS under 3.7.11, Control Room Emergency Air Temperature Control System (CREATCS). The requirement of the associated Surveillance Requirement (SR) 3.7.11 is to demonstrate the ability of each train to remove its assumed heat load.

Summary of Evaluation:

The compensatory actions involve manually throttling valve 0-THV-067-0623A when the MCR Chiller A is in service. It is necessary to adjust the ERCW flow through the condenser due to changes in the ERCW temperature. In addition it is necessary to close the valve when the

chiller is out of service. Closing the valve when MCR Chiller A is out of service will require the chiller to be declared inoperable due to unavailability of cooling water to the chiller condenser.

34. Evaluation: CR 1092054, Evaluation R0

Activity Description:

The proposed change is an Accept-As-Is evaluation for CR 1092054. The condition being evaluated is the possibility that a 1/4-20x1/2" stainless steel cap screw may be in the upper reactor internals when Unit 1 starts up from RFO13.

CR 1092054 documents that while R.O.V. Technologies was using their crawler to vacuum the reactor cavity rust ring, the rotating brush stopped working. When the crawler was retrieved, two cap screws were discovered to be missing from the bracket holding the brush. One screw was recovered immediately. The second cap screw could not be located and must be assumed to be in the Reactor Pressure Vessel (RPV).

The dimensions of the cap screw were determined from R.O.V. Technologies assembly drawing for the cleaning tool from which the cap screw was lost, and McMaster-Carr drawing/part number 92196A537 which depicts the socket head cap screw. The cap screw is 0.75 inch long, with a 0.25 inch long, 0.375-inch diameter head and 0.5 inch of threaded (1/4" diameter, 20 threads per inch) shank. Based on these dimensions, the weight of the cap screw is approximately 0.015 lb.

At the time the screw was lost, the reactor upper internals were installed. Therefore, if the screw is inside the RPV, it must be assumed that it is on top of the upper internals at the start of Unit 1 Cycle 14. Based on the Westinghouse evaluation performed for this issue no adverse impacts to the reactor vessel internal structure, the vessel itself, or other components in the RCS system are anticipated either short term or in the long term.

Summary of Evaluation:

The Westinghouse evaluation of the effects of the cap screw in the RCS during operation determined that, although there are ways in which the screw could induce detrimental effects, the likelihood of an accident or malfunction is not increased as such magnitude of detrimental impact is not plausible. The Westinghouse evaluation expects no more-than-minimal increase in the consequences of any accident or malfunction because both cases are bound by the pre-existing assumptions for failed fuel (i.e., all of the rods in the assembly are assumed damaged) and for the highest-worth RCCA being completely retracted. The cap screw is too small to induce, or cause other debris that could induce, RCS inventory loss or component damage such that a new type of accident or malfunction is created. Impact and cumulative wear from the screw will not produce damage to the fuel cladding or RCS pressure boundary beyond the bounds of typical operational leakage.

35. Evaluation: CR 1127526, Evaluation R0

Activity Description:

The proposed change is an Accept-As-Is evaluation for CR 1127526. The condition being evaluated is the possibility that ball bearings from the two SQUID (SUPREEM™ eQuivalent safe-end Ultrasonic Inspection Device) tools used by WesDyne International used during Unit 1 Refueling Outage 13 in the Fall of 2015 may be in the Unit 1 RCS. CR 1127526 documents a

generic operability assessment for Westinghouse-designed Nuclear Steam supply System (NSSS) plants that are potentially affected by ball bearings that were found to be missing from two SQUID tools used by WesDyne International to perform remote inspections for nozzle safe-end welds. One of the SQUID tools is known to have discharged 0.09375-inch diameter, chrome plated ball bearings in a reactor vessel nozzle during inspections at a plant in October 2015. Another similar SQUID tool was found to be missing 14 ball bearings when it was examined at the Westinghouse Waltz Mill site after it was returned from use at another plant during the fall 2015 outage season.

WesDyne has concluded that it cannot be determined whether the ball bearings in either of the SQUID tools were all present before their use during the fall 2015 outage season. Therefore, WesDyne has conservatively assumed that 14-25 ball bearings could have been lost from the tools at any point because the tools were placed in service. These ball bearings measure (nominally) 0.09375 in diameter, and are made of chrome plated, carbon steel.

Summary of Evaluation:

The Westinghouse evaluation of the effects of the ball bearings in the RCS during operation determined that although there are ways in which the ball bearings could induce detrimental effects, the likelihood of an accident or malfunction is not increased as such magnitude of detrimental impact is not plausible. The evaluation expects no more than minimal increase in the consequences of any accident or malfunction because both are bound by the pre-existing assumptions for failed fuel and for the highest-worth RCCA being completely retracted. The ball bearings are too small to induce, or cause other debris that could induce RCS inventory loss or component damage such that a new type of accident or malfunction is created. Impact and wear from the ball bearings will not produce damage to the fuel cladding or RCS pressure boundary beyond the bounds of typical operational leakage.

36. Evaluation: FSAR Change 13-020, Evaluation R0

Activity Description:

This evaluation reviews the changes being made to Unit 1 UFSAR Table 15.5.9, Doses From Loss-of-Coolant Accident due to the resolution of CRs 913900 (Unit 1) and 883559 (Unit 2). (Unit 2 was evaluated separately.)

The Unit 1 CR 913900 was initiated to track the required Unit 1 corrective actions to completion for the condition documented in existing Unit 2 CR 883559. The Condition Reports involve the Emergency Gas Treatment System (EGTS) which is shared by both Unit 1 and Unit 2.

The CR documents that the incorrect EGTS flow rate was used in source term files for all dual train EGTS cases in calculation TIRPS197 Rev. 22, Offsite Doses Due to a Regulatory Guide 1.4 Loss of Coolant Accident. The correct flow rate in calculation TIRPS197 Rev. 23 resulted in a slight increase in offsite dose numbers for Unit 1 and Unit 2.

Calculation TIRPS197 provided input values to calculation TI-RPS-198 (TIRPS198), Revision 24, Dose to Control Room Personnel Due to a Regulatory Guide 1.4 Loss of Coolant Accident. Calculation TIRPS198, Revision 25 resulted in a slight increase in offsite dose numbers for Unit 1 and Unit 2.

The increases in dose levels have been evaluated. The evaluation demonstrates that the changes in dose rates are minimal when using the correct air flow rate.

The EGTS is provided for ventilation control and cleanup of the atmosphere inside the annulus between the Shield Building and the Primary Containment Building. It is shared by both units. This system has two subsystems; the annulus vacuum control subsystem and the air cleanup subsystem.

The overall design bases for the EGTS are:

1. To keep the air pressure within each Shield Building annulus below atmospheric pressure at all times in which the integrity of that particular containment is required.
2. To reduce the concentration of radioactive nuclides in annulus air that is released to the environs during a LOCA in either reactor unit to levels sufficiently low to keep the site boundary and low population zone (LPZ) dose rates below the 10 CFR 100 values.
3. To withstand the safe shutdown earthquake.
4. To provide for initial and periodic testing of the system capability to function as designed.

The design bases of the EGTS Air Cleanup Units are:

1. To provide fission product removal capabilities sufficient to keep radioactivity levels in the Shield Building annulus air released to the environs during a DBA LOCA sufficiently low to assure compliance with 10 CFR 100 guidelines.

Summary of Evaluation:

The calculations TIRPS197 and TIRPS198 evaluate the consequences of a LOCA. This evaluation determined that the increase in consequences when the error in the flow rate was corrected was minimal. There are no physical modifications or changes in the manner in which the plant will be operated as a result of this UFSAR Change.

37. Evaluation: SAR Change Package No. U2-016, Evaluation R0

Activity Description:

The proposed activity revises the Unit 2 UFSAR to acknowledge the use of the PARAGON/NEXUS methodology as a replacement for the PHOENIX lattice physics code. The PARAGON/NEXUS methods were approved by the NRC as a replacement for PHOENIX. The PARAGON code improves the method used to solve for the neutron flux (transport solution).

Summary of Evaluation:

A 50.59 evaluation is required because the method of evaluation described in UFSAR Chapter 4 (PHOENIX) is being revised to acknowledge the use of PARAGON/NEXUS. 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 PARAGON/NEXUS does not constitute a departure because (1) the methodology is approved by the NRC specifically for PWR uranium fueled cores and (2) the application of these codes for Unit 2 is consistent with the terms, conditions, and limitations of that NRC approval.

38. Evaluation: IBCR-14-067-3 Rev. 4, Evaluation R3

Activity Description:

This Interface Boundary Clearance Release (IBCR) for the ERCW 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 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 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.

This evaluation also considers the impact caused by removing power from the open ERCW supply valve to the Station Air Compressor on the train not being tested while the tested train's ERCW supply valve is isolated. Removing power from this valve helps ensure it will not spuriously close and lead to a plant trip (caused by a loss of the Station Air Compressors). The ERCW supply valves to the Station Air Compressors, however, are designed to close upon receipt of high header flow or low pressure. Removing power from these valves will prevent this automatic closure function. A dedicated manual operator will be stationed at the valve's MCC and will restore power at the initiation of any non-normal event or accident. A standing closure signal will close the valve (if present) as soon as power is restored to valve. Closure of the ERCW supply valve to the Station Air Compressors for the tested train puts the valve in its safety related position. The Station Air Compressors serve no safety related function and are not required for accident mitigation.

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.

The conclusion is that the change can be implemented per existing processes without obtaining a licensing amendment.

39. Evaluation: IBCR-15-211-002 Rev. 0, Evaluation R0

Activity Description:

IBCR-15-211-002 permanently connects Emergency Diesel Generator (EDG) 2A-A and 2B-B load sequencing timers for the unit 2 train A and train B motor driven AFW pumps, Charging pumps, Safety Injection pumps, Containment Spray pumps, RHR pumps, and Pressurizer Backup Heaters in accordance with 0-TI-12.08. The purpose of the timers is to allow EDG voltage and frequency to recover so the EDG will be capable starting and powering subsequent loads. All four EDGs are the same design, the unit 2 loads to be applied are similar to unit 1, and the sequence timers have the same set points. An Additional Requirement requires successful preoperational testing of one train prior to testing the opposite train.

Summary of Evaluation:

The proposed configuration does 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 Interface Boundary Clearance Release.

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.