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

September 9, 2010

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

10 CFR 50.4

Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

Subject: **WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING FINAL SAFETY ANALYSIS REPORT (FSAR) RELATED TO INSTRUMENTATION AND CONTROLS (TAC NO. ME2371)**

References:
1. NRC letter dated August 25, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Final Safety Analysis Report Related to Instrumentation and Controls (TAC NO. ME2371)"
2. TVA letter dated February 8, 2008, "Watts Bar Nuclear Plant, Unit 2 - Final Safety Analysis Report (FSAR) Red-Line For Unit 2"

The purpose of this letter is to provide additional information requested by NRC in Reference 1 in support of its review of WBN's Unit 2 FSAR, Chapter 7, "Instrumentation and Controls." Specifically, NRC requested TVA to:

"Provide a complete listing and summary description of all significant design changes made to Unit 1 that did not receive prior NRC staff approval under 10 CFR 50.59 affecting the instrumentation and controls systems discussed in FSAR Chapter 7. TVA should also provide (1) a description of the criteria it used to identify the significant changes and (2) a table correlating each 10 CFR 50.59 evaluation to the affected FSAR subsections."

TVA is providing a list of changes to the Unit 1 UFSAR contained in Amendments 0 through 8. A change was identified if it required more than a 10 CFR 50.59 screening review and:

- Made a physical modification to the plant
- Changed a program
- Changed a process
- Changed an analysis

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Enclosure 1 provides a summary (by UFSAR section number) of the changes meeting the previous criteria (Reference 2) that resulted in a change to the content/verbiage in Unit 1 UFSAR Chapter 7 as contained in Amendments 0 through 8. This listing correlates each 10 CFR 50.59 evaluation to the affected UFSAR subsection.

Enclosure 2 provides a summary of other 10 CFR 50.59 evaluations not meeting the Enclosure 1 criteria.

Enclosure 3 provides a list of Unit 1 UFSAR Chapter 7 changes which required a license amendment.

There are no new regulatory commitments contained in this letter. If you have any questions, please contact William Crouch at (423) 365-2004.

I declare under the penalty of perjury that the foregoing is true and correct. Executed on the 9th day of September 2010.

Sincerely,

M. Bajestani

Masoud Bajestani
Watts Bar Unit 2 Vice President

cc (Enclosures):

1. Summary Listing and Description of the Changes Resulting in a Change to the Content/Verbiage in Unit 1 UFSAR Chapter 7 as Contained in Amendments 0 through 8

2. Summary Listing and Description of the Other 10 CFR 50.59 Evaluations

3. List of Unit 1 UFSAR Chapter 7 Changes Which Required a License Amendment

U. S. Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Ave., NE Suite 1200
Atlanta, Georgia 30303-1257

NRC Resident Inspector Unit 2
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

ENCLOSURE 1

TVA LETTER DATED SEPTEMBER 9, 2010
WATTS BAR NUCLEAR PLANT (WBN) UNIT 2
REQUEST FOR ADDITIONAL INFORMATION REGARDING FSAR
RELATED TO INSTRUMENTATION AND CONTROLS (TAC NO. ME2371)

SUMMARY LISTING AND DESCRIPTION OF THE CHANGES
RESULTING IN A CHANGE TO THE CONTENT/VERBIAGE IN THE UNIT 1 UFSAR
CHAPTER 7 AS CONTAINED IN AMENDMENTS 0 THROUGH 8

SUMMARY LISTING AND DESCRIPTION OF THE CHANGES
RESULTING IN A CHANGE TO THE CONTENT/VERBIAGE IN THE UNIT 1 UFSAR
CHAPTER 7 AS CONTAINED IN AMENDMENTS 0 THROUGH 8

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
0	7.1	WBPLEE-97-089-0 M-39293-A 1473	This Safety Assessment/Evaluation addresses changes in Cycle 2 design basis analyses parameters and changes in instrumentation response and control characteristics and setpoints as discussed in Subsections a & b. These changes provide more operational margin as well as operational enhancements in reactor control and protection. The design basis analyses which support these changes also satisfy commitments made to the NRC to provide consistency between analysis parameters used for LOCA and non-LOCA analyses in the reactor vessel up flow conversion (WCAP-11696 and WCAP-11627) and to address 10CFR50.46 modeling issues. The respective TVA letters to NRC are dated July 28,1993 "Watts Bar Nuclear Plant (\NBN) - Emergency Core Cooling System (ECCS) -Evaluation Model Changes (TAC Nos M86069 and M86070)", LBLOCA NCO 920041002 &-03 and letter dated August 28,1995 "Watts Bar Nuclear Plant (WBN)Emergency Core Cooling System (ECCS) -Evaluation Model Changes (TAC Nos M86069 and M86070)", SBLOCA NCO 920041006. In addition, the Cycle 2 design basis analyses parameters account for a longer fuel cycle (greater than 12 months but less than 18 months). Note that the changes associated with the 18 month fuel cycle are covered in a separate Safety Assessment/Evaluation. Since the safety analysis input parameter changes associated with the 18 month fuel cycle are included in the analyses which support this safety evaluation, the SA/SE of the changes associated with the 18 month fuel cycle can be performed separately without affecting the conclusions of this SA/SE. The safety analysis assumptions and input parameter changes and the various plant hardware setpoint changes evaluated in this SA/SE are implemented via DCN M-39293-A, FSAR Change Package 1473, and TRM Change package 97-002 and include: a. Setpoint/Scaling Changes-DCN M-39293-A • For Cycle 2 and later revise the OT-T/OP-T reactor trip setpoints by adjusting gains

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			and lead/lag time constants to enhance operating margin and incorporate tolerances for variations in indicated loop delta T and Tavg in the ΔT/Tavg and Low SG level Trip Time Delay (TID) channels. • Revise setpoint and scaling uncertainty analyses to support Cycle 2 and later operation with 10% SGTP and additional 2% RTDF and provide the basis for a TS change to allow operation at reduced flow, if it becomes necessary to take advantage of the additional margin. • Revise the control system power mismatch non-linear gain breakpoint from 1 0A» to 2% to further minimize rod stepping due to power oscillations which can be caused by lower plenum anomaly. • Revise Pressurizer Pressure SSDs to revise compliance value for DNB limit. • Revise SG Level Control Point from 66.5% down to 60% to reduce moisture carryover. Some of the changes require modification to the plant Technical Specifications and have been addressed in TS Change Package 96-013 as submitted to the NRC (see discussion relative to these in the following text and Reference 2q). In addition the issues of Westinghouse Technical Bulletin ESBU-TB-96-07 and the WBN response are addressed. This bulletin addresses the impact of hot and cold leg streaming and changes in hot leg streaming due to burnup-dependent radial power redistribution on indicated delta T (delta To) delta T span, and the uncertainty calculations for the OT delta T and OP delta T reactor trip functions. The WBN response to this bulletin committed to implement bulletin recommendations, including verification of the analysis limits for hot and cold leg temperatures and evaluating the effects of

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			operation at reduced full power loop delta T values on the OT delta T and OP delta T uncertainty allowances. b. Analyses Changes-FSAR Change Package1473 and TRM Change Package 97-002 1. Steam Generator Tube Plugging (SGTP) and Reduction in Thermal Design Flow (RTDF) 2. Rod Control System Optimization 3. Reduced Feedwater Temperature at 0 to 25% Power 4. Increased Spray Initiation Delay Time for Containment Integrity 5. Increased OT-T/OP-T Response Time from 7 to 8 Seconds 6. Reduced Steam Generator Water Level from 66.5% to 60% 7. OT delta T and OP delta T Reactor Trip Setpoints Margin Enhancement c. LOCA and SGTR Analyses 1. Large Loss of Coolant Accident (LBLOCA) 2. Small Break Loss of Coolant Accident (SBLOCA) 3. Steam Generator Tube Rupture (SGTR) d Non-LOCA Related Analyses: The Non-LOCA analyses have been performed for the following plant changes. Cycle 2 design features (e.g. - F_Q = 2.5, $F_{\Delta H}$ = 1.60)* • Steam generator tube plugging (SGTP) of 10% • Reduced thermal design flow (RTDF) of 2% • OT delta T/OP delta T margin enhancement • OT delta T/OP delta T response time increase

SUMMARY LISTING AND DESCRIPTION OF THE CHANGES RESULTING IN A CHANGE TO THE CONTENT/VERBIAGE IN THE UNIT 1 UFSAR CHAPTER 7 AS CONTAINED IN AMENDMENTS 0 THROUGH 8

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			• Rod Control System Optimization (revised setpoints) • Reduced feedwater temperature at 0 to 25% power • Reduced steam generator water level at 100% power The setpoint/scaling changes do not increase the probability of an accident. The proposed changes do not result in a condition where the design material and construction standards, which were applicable prior to the changes, are altered. The OT delta T and OP delta T setpoints/scaling changes do not require any physical hardware changes and are used in the various defined analyses for accident definition/mitigation. All of the affected NSSS systems and components have been evaluated with the Total Design Flow (TDF) of 93,100 gpm. The primary loop components (reactor vessel. reactor internals, CRDMs, loop piping and supports, reactor coolant pump, steam generator. and pressurizer) meet the applicable structural limits with the revised TDF of 93.100 gpm and will continue to perform their design functions. The RCCA drop time remains unaffected and the current design core bypass flow remains valid. All of the applicable acceptance criteria for the accidents described in the FSAR continue to be met. There is no increase in the consequences of an accident: The SLB radiological doses are unaffected and are still within the existing licensing basis limits. The proposed changes do not cause an increase in the probability of an accident. The changes in the tolerances for delta TO, T' and T" also do not require any physical hardware modifications and only require changes to the Technical Specification Allowable Values for the OP delta T and OT delta T setpoints and for the vessel delta T equivalent to power functions. Thus, there is no increase in the probability of an accident

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			since the appropriate Allowable Values have been modified to determine channel operability for these functions. The proposed changes do not cause the initiation of any accident nor create any new limiting single failures. The OT delta T and OP delta T protection functions are used for accident mitigation and do not initiate any accidents. Also, the affected systems and components will still perform their intended design functions. The proposed changes do not create any new failure modes for safety related equipment. The changes do not result in any original design specification, such as seismic requirements, electrical separation requirements or equipment qualification being altered. The margin of safety for the applicable safety analyses has not been reduced. All of the applicable DNB limits continue to be met for the non-LOCA analyses. The LBLOCA input parameters do not require adjustment for the TDF of 93.100 gpm. The SBLOCA has been reanalyzed for the TDF of 93,100 gpm, and the SBLOCA PCT is well below the 2200°F limit. The affected NSSS systems and components will still meet the applicable design limits and perform their intended safety functions with the TDF of 93,100 gpm. Also, the SLB and LOCA Mass and Energy releases are still within the applicable equipment qualification limits. The SGTR doses remain within the applicable 10 CFR 100 limits, and the steam generator margin to overfill is maintained.
0	7.2	WBPLEE-97-089-0 M-39293-A 1473	This Safety Assessment/Evaluation addresses changes in Cycle 2 design basis analyses parameters and changes in instrumentation response and control characteristics and setpoints as discussed in Subsections a & b. These changes provide more operational margin as well as operational enhancements in reactor control and protection. The design basis analyses which support these changes also satisfy commitments made to the NRC to provide consistency between analysis parameters used for LOCA and non-LOCA analyses in the reactor vessel up flow conversion (WCAP-11696 and WCAP-

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			11627) and to address 10CFR50.46 modeling issues. The respective TVA letters to NRC are dated July 28,1993 "Watts Bar Nuclear Plant (\NBN) - Emergency Core Cooling System (ECCS) -Evaluation Model Changes (TAC Nos M86069 and M86070)", LBLOCA NCO 920041002 &-03 and letter dated August 28,1995 "Watts Bar Nuclear Plant (WBN)Emergency Core Cooling System (ECCS) -Evaluation Model Changes (TAC Nos M86069 and M86070)", SBLOCA NCO 920041006. In addition, the Cycle 2 design basis analyses parameters account for a longer fuel cycle (greater than 12 months but less than 18 months). Note that the changes associated with the 18 month fuel cycle are covered in a separate Safety Assessment/Evaluation. Since the safety analysis input parameter changes associated with the 18 month fuel cycle are included in the analyses which support this safety evaluation, the SA/SE of the changes associated with the 18 month fuel cycle can be performed separately without affecting the conclusions of this SA/SE. The safety analysis assumptions and input parameter changes and the various plant hardware setpoint changes evaluated in this SA/SE are implemented via DCN M-39293-A, FSAR Change Package 1473, and TRM Change package 97-002 and include: a. Setpoint/Scaling Changes-DCN M-39293-A • For Cycle 2 and later revise the OT-T/OP-T reactor trip setpoints by adjusting gains and lead/lag time constants to enhance operating margin and incorporate tolerances for variations in indicated loop delta T and Tavg in the ΔT/Tavg and Low SG level Trip Time Delay (TID) channels. • Revise setpoint and scaling uncertainty analyses to support Cycle 2 and later operation with 10% SGTP and additional 2% RTDF and provide the basis for a TS change to allow operation at reduced flow, if it becomes necessary to take advantage

SUMMARY LISTING AND DESCRIPTION OF THE CHANGES RESULTING IN A CHANGE TO THE CONTENT/VERBIAGE IN THE UNIT 1 UFSAR CHAPTER 7 AS CONTAINED IN AMENDMENTS 0 THROUGH 8

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			of the additional margin. • Revise the control system power mismatch non-linear gain breakpoint from 1 0A» to 2% to further minimize rod stepping due to power oscillations which can be caused by lower plenum anomaly. • Revise Pressurizer Pressure SSDs to revise compliance value for DNB limit. • Revise SG Level Control Point from 66.5% down to 60% to reduce moisture carryover. Some of the changes require modification to the plant Technical Specifications and have been addressed in TS Change Package 96-013 as submitted to the NRC (see discussion relative to these in the following text and Reference 2q). In addition the issues of Westinghouse Technical Bulletin ESBU-TB-96-07 and the WBN response are addressed. This bulletin addresses the impact of hot and cold leg streaming and changes in hot leg streaming due to burn up-dependent radial power redistribution on indicated delta T (delta To) delta T span, and the uncertainty calculations for the OT delta T and OP delta T reactor trip functions. The WBN response to this bulletin committed to implement bulletin recommendations, including verification of the analysis limits for hot and cold leg temperatures and evaluating the effects of operation at reduced full power loop delta T values on the OT delta T and OP delta T uncertainty allowances. b. Analyses Changes-FSAR Change Package1473 and TRM Change Package 97-002. 1. Steam Generator Tube Plugging (SGTP) and Reduction in Thermal Design Flow (RTDF)

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			2. Rod Control System Optimization 3. Reduced Feedwater Temperature at 0 to 25% Power 4. Increased Spray Initiation Delay Time for Containment Integrity 5. Increased OT-T/OP-T Response Time from 7 to 8 Seconds 6. Reduced Steam Generator Water Level from 66.5% to 60% 7. OT delta T and OP delta T Reactor Trip Setpoints Margin Enhancement c. LOCA and SGTR Analyses 1. Large Loss of Coolant Accident (LBLOCA) 2. Small Break Loss of Coolant Accident (SBLOCA) 3. Steam Generator Tube Rupture (SGTR) d Non-LOCA Related Analyses: The Non-LOCA analyses have been performed for the following plant changes. Cycle 2 design features (e.g. - F_Q = 2.5, $F_{\Delta H}$ = 1.60)* • Steam generator tube plugging (SGTP) of 10% • Reduced thermal design flow (RTDF) of 2% • OT delta T/OP delta T margin enhancement • OT delta T/OP delta T response time increase • Rod Control System Optimization (revised setpoints) • Reduced feedwater temperature at 0 to 25% power • Reduced steam generator water level at 100% power The setpoint/scaling changes do not increase the probability of an accident. The proposed changes do not result in a condition where the design material and

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			construction standards, which were applicable prior to the changes, are altered. The OT delta T and OP delta T setpoints/scaling changes do not require any physical hardware changes and are used in the various defined analyses for accident definition/mitigation. All of the affected NSSS systems and components have been evaluated with the Total Design Flow (TDF) of 93,100 gpm. The primary loop components (reactor vessel. reactor internals, CRDMs, loop piping and supports, reactor coolant pump, steam generator. and pressurizer) meet the applicable structural limits with the revised TDF of 93.100 gpm and will continue to perform their design functions. The RCCA drop time remains unaffected and the current design core bypass flow remains valid. All of the applicable acceptance criteria for the accidents described in the FSAR continue to be met. There is no increase in the consequences of an accident: The SLB radiological doses are unaffected and are still within the existing licensing basis limits. The proposed changes do not cause an increase in the probability of an accident. The changes in the tolerances for delta TO, T' and T" also do not require any physical hardware modifications and only require changes to the Technical Specification Allowable Values for the OP delta T and OT delta T setpoints and for the vessel delta T equivalent to power functions. Thus, there is no increase in the probability of an accident since the appropriate Allowable Values have been modified to determine channel operability for these functions. The proposed changes do not cause the initiation of any accident nor create any new limiting single failures. The OT delta T and OP delta T protection functions are used for accident mitigation and do not initiate any accidents. Also, the affected systems and components will still perform their intended design functions.

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			The proposed changes do not create any new failure modes for safety related equipment. The changes do not result in any original design specification, such as seismic requirements, electrical separation requirements or equipment qualification being altered. The margin of safety for the applicable safety analyses has not been reduced. All of the applicable DNB limits continue to be met for the non-LOCA analyses. The LBLOCA input parameters do not require adjustment for the TDF of 93.100 gpm. The SBLOCA has been reanalyzed for the TDF of 93,100 gpm, and the SBLOCA PCT is well below the 2200°F limit. The affected NSSS systems and components will still meet the applicable design limits and perform their intended safety functions with the TDF of 93,100 gpm. Also, the SLB and LOCA Mass and Energy releases are still within the applicable equipment qualification limits. The SGTR doses remain within the applicable 10 CFR 100 limits, and the steam generator margin to overfill is maintained.
3	7.2	WBPLMN-01-025-0 E-50952-A 1676	This document change provides for the disabling of the P-12 interlock which thereby provides an alternate method of using additional condenser dump valves for unit cool down. At present, the steam dump logic will block the condenser dump valves when the plant TAVG is reduced below the low-low TAVG interlock (P-12); value of 550°F. A manual interlock bypass switch is provided to permit the use of the designated cool down valves. These cool down valves are three out of the total of twelve condenser dump valves. The P-12 interlock will be disabled by lifting wires for the K631 relay contacts associated with the two independent steam dump control circuits. The disablement will be performed procedurally with no permanent hardware modifications to the unit. The use of additional condenser dump valves will be optional for the Operator. The condenser dump valves are controlled using the steam pressure controller before and after the P-12 interlock is disabled. This procedurally controlled temporary alteration will allow the use of additional condenser dump valves below 350°F to aid in the cool down of the RCS during unit cool down.

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			Normal cool down now uses all twelve condenser dump valves as needed above TAVG temperature of 550°F and the three cool down valves below this temperature value. The cool down valves are capable of being operated below 550°F by using the MCR hand switches, which disable the automatic blocking performed by the P-12 interlock for these 3 condenser dump valves. The effectiveness of the cool down valves for unit cool down decreases as RCS temperature and pressure decrease. The proposed method provides for disabling the P-12 interlock and permits the use of additional condenser dump valves during cool down below 350°F which is the Mode 4 entry temperature. There is not an increase in the frequency of occurrence of an accident or malfunction of a component important to safety previously evaluated in the UFSAR. Overcooling events involving increased heat removal by the secondary system are analyzed in Chapter 15 of the UFSAR. These include "Excessive Heat Removal Due to Feedwater System Malfunctions" associated with a rapid increase in steam flow, "Excessive Load Increase Incident" associated with a rapid increase in steam flow, "Accidental Depressurization of the Main Steam System" associated with an inadvertent opening of a single condenser dump, relief or safety valve, and "Major Rupture of a Main Steam Line." Since the reactor is shutdown, Mode 4 shutdown margin will be assured by increasing boron concentration (i.e., Mode 4 at 200.1°F concentration) prior to allowing additional condenser dump valves overcooling event involving return to criticality is not credible at this phase of shutdown operation. The RHR system is available as before to prevent uncontrolled heat up of the RCS. The steam generator power operated relief valves are also available as a means of primary cooling via the secondary main steam system. Thus, the probability of an uncontrolled heat up due to failure of a cooling system or component (e.g., steam dumps) is not increased.

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			The consequences of the accidents and malfunctions of the associated equipment remains unchanged whether the three cool down valves or the additional steam dump valves are used. The potential to create a new type of event not previously evaluated in the UFSAR is not found with this change. No new accidents are created by this change and no new control features have been incorporated. The disablement of the P-12 interlock will be performed procedurally with no permanent hardware modifications to the unit. The use of this cool down method does not affect the design basis limit of any fission product barriers. The rate of cool down is maintained within the acceptance limits for RCS cool down, as specified in the technical specifications (which ensure compliance with 10 CFR 50 Appendix G). The plant cool down is controlled in the same manner as previously considered which is manual operator action. The method of evaluation for the identified accidents is not modified nor is a new method created. This change does not have an impact on evaluation methodologies described in the UFSAR. Therefore, this change does not require a license amendment.
0	7.3	WBPLEE-97-089-0 M-39293-A 1473	This Safety Assessment/Evaluation addresses changes in Cycle 2 design basis analyses parameters and changes in instrumentation response and control characteristics and setpoints as discussed in Subsections a & b. These changes provide more operational margin as well as operational enhancements in reactor control and protection. The design basis analyses which support these changes also satisfy commitments made to the NRC to provide consistency between analysis parameters used for LOCA and non-LOCA analyses in the reactor vessel up flow conversion (WCAP-11696 and WCAP-11627) and to address 10CFR50.46 modeling issues. The respective TVA letters to NRC are dated July 28,1993 "Watts Bar Nuclear Plant (\NBN) - Emergency Core Cooling System (ECCS) -Evaluation Model Changes (TAC Nos M86069 and M86070)", LBLOCA NCO 920041002 &-03 and letter dated August 28,1995 "Watts Bar Nuclear Plant (WBN)Emergency Core Cooling System (ECCS) -Evaluation Model Changes (TAC Nos M86069 and M86070)", SBLOCA NCO 920041006. In addition, the Cycle 2 design basis analyses parameters account for a longer fuel cycle

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			(greater than 12 months but less than 18 months). Note that the changes associated with the 18 month fuel cycle are covered in a separate Safety Assessment/Evaluation. Since the safety analysis input parameter changes associated with the 18 month fuel cycle are included in the analyses which support this safety evaluation, the SA/SE of the changes associated with the 18 month fuel cycle can be performed separately without affecting the conclusions of this SA/SE. The safety analysis assumptions and input parameter changes and the various plant hardware setpoint changes evaluated in this SA/SE are implemented via DCN M-39293-A, FSAR Change Package 1473, and TRM Change package 97-002 and include: a. Setpoint/Scaling Changes-DCN M-39293-A • For Cycle 2 and later revise the OT-T/OP-T reactor trip setpoints by adjusting gains and lead/lag time constants to enhance operating margin and incorporate tolerances for variations in indicated loop delta T and Tavg in the ΔT/Tavg and Low SG level Trip Time Delay (TID) channels. • Revise setpoint and scaling uncertainty analyses to support Cycle 2 and later operation with 10% SGTP and additional 2% RTDF and provide the basis for a TS change to allow operation at reduced flow, if it becomes necessary to take advantage of the additional margin. • Revise the control system power mismatch non-linear gain breakpoint from 1 0A» to 2% to further minimize rod stepping due to power oscillations which can be caused by lower plenum anomaly. • Revise Pressurizer Pressure SSDs to revise compliance value for DNB limit.

<table>
<tr><th>Unit 1 UFSAR Amendment No</th><th>Section No.</th><th>Safety Evaluation No./
Design Change No./
UFSAR Change Package No.</th><th>Summary Description</th></tr>
<tr><td></td><td></td><td></td><td>• Revise SG Level Control Point from 66.5% down to 60% to reduce moisture carryover.

Some of the changes require modification to the plant Technical Specifications and have been addressed in TS Change Package 96-013 as submitted to the NRC (see discussion relative to these in the following text and Reference 2q).

In addition the issues of Westinghouse Technical Bulletin ESBU-TB-96-07 and the WBN response are addressed. This bulletin addresses the impact of hot and cold leg streaming and changes in hot leg streaming due to burn up-dependent radial power redistribution on indicated delta T (delta To) delta T span, and the uncertainty calculations for the OT delta T and OP delta T reactor trip functions. The WBN response to this bulletin committed to implement bulletin recommendations, including verification of the analysis limits for hot and cold leg temperatures and evaluating the effects of operation at reduced full power loop delta T values on the OT delta T and OP delta T uncertainty allowances.

b. Analyses Changes-FSAR Change Package1473 and TRM Change Package 97-002

1. Steam Generator Tube Plugging (SGTP) and Reduction in Thermal Design Flow (RTDF)
2. Rod Control System Optimization
3. Reduced Feedwater Temperature at 0 to 25% Power
4. Increased Spray Initiation Delay Time for Containment Integrity
5. Increased OT-T/OP-T Response Time from 7 to 8 Seconds
6. Reduced Steam Generator Water Level from 66.5% to 60%
7. OT delta T and OP delta T Reactor Trip Setpoints Margin Enhancement

c. LOCA and SGTR Analyses</td></tr>
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Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			1. Large Loss of Coolant Accident (LBLOCA) 2. Small Break Loss of Coolant Accident (SBLOCA) 3. Steam Generator Tube Rupture (SGTR) d Non-LOCA Related Analyses: The Non-LOCA analyses have been performed for the following plant changes. Cycle 2 design features (e.g. - F_Q = 2.5, $F_{\Delta H}$ = 1.60)* • Steam generator tube plugging (SGTP) of 10% • Reduced thermal design flow (RTDF) of 2% • OT delta T/OP delta T margin enhancement • OT delta T/OP delta T response time increase • Rod Control System Optimization (revised setpoints) • Reduced feedwater temperature at 0 to 25% power • Reduced steam generator water level at 100% power The setpoint/scaling changes do not increase the probability of an accident. The proposed changes do not result in a condition where the design material and construction standards, which were applicable prior to the changes, are altered. The OT delta T and OP delta T setpoints/scaling changes do not require any physical hardware changes and are used in the various defined analyses for accident definition/mitigation. All of the affected NSSS systems and components have been evaluated with the Total Design Flow (TDF) of 93,100 gpm. The primary loop components (reactor vessel. reactor internals, CRDMs, loop piping and supports, reactor coolant pump, steam generator. and pressurizer) meet the applicable structural limits with the revised TDF of

Unit 1 UFSAR Amendment No	Section No.	Safety Evaluation No./ Design Change No./ UFSAR Change Package No.	Summary Description
			93.100 gpm and will continue to perform their design functions. The RCCA drop time remains unaffected and the current design core bypass flow remains valid. All of the applicable acceptance criteria for the accidents described in the FSAR continue to be met. There is no increase in the consequences of an accident: The SLB radiological doses are unaffected and are still within the existing licensing basis limits. The proposed changes do not cause an increase in the probability of an accident. The changes in the tolerances for delta TO, T' and T" also do not require any physical hardware modifications and only require changes to the Technical Specification Allowable Values for the OP delta T and OT delta T setpoints and for the vessel delta T equivalent to power functions. Thus, there is no increase in the probability of an accident since the appropriate Allowable Values have been modified to determine channel operability for these functions. The proposed changes do not cause the initiation of any accident nor create any new limiting single failures. The OT delta T and OP delta T protection functions are used for accident mitigation and do not initiate any accidents. Also, the affected systems and components will still perform their intended design functions. The proposed changes do not create any new failure modes for safety related equipment. The changes do not result in any original design specification, such as seismic requirements, electrical separation requirements or equipment qualification being altered. The margin of safety for the applicable safety analyses has not been reduced. All of the applicable DNB limits continue to be met for the non-LOCA analyses. The LBLOCA

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			input parameters do not require adjustment for the TDF of 93.100 gpm. The SBLOCA has been reanalyzed for the TDF of 93,100 gpm, and the SBLOCA PCT is well below the 2200°F limit. The affected NSSS systems and components will still meet the applicable design limits and perform their intended safety functions with the TDF of 93,100 gpm. Also, the SLB and LOCA Mass and Energy releases are still within the applicable equipment qualification limits. The SGTR doses remain within the applicable 10 CFR 100 limits, and the steam generator margin to overfill is maintained.
3	7.3	WBPLMN-01-025-0 E-50952-A 1676	This document change provides for the disabling of the P-12 interlock which thereby provides an alternate method of using additional condenser dump valves for unit cool down. At present, the steam dump logic will block the condenser dump valves when the plant TAVG is reduced below the low-low TAVG interlock (P-12); value of 550°F. A manual interlock bypass switch is provided to permit the use of the designated cool down valves. These cool down valves are three out of the total of twelve condenser dump valves. The P-12 interlock will be disabled by lifting wires for the K631 relay contacts associated with the two independent steam dump control circuits. The disablement will be performed procedurally with no permanent hardware modifications to the unit. The use of additional condenser dump valves will be optional for the Operator. The condenser dump valves are controlled using the steam pressure controller before and after the P-12 interlock is disabled. This procedurally controlled temporary alteration will allow the use of additional condenser dump valves below 350°F to aid in the cool down of the RCS during unit cool down. Normal cool down now uses all twelve condenser dump valves as needed above TAVG temperature of 550°F and the three cool down valves below this temperature value. The cool down valves are capable of being operated below 550°F by using the MCR hand switches, which disable the automatic blocking performed by the P-12 interlock for these 3 condenser dump valves. The effectiveness of the cool down valves for unit cool down decreases as RCS

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			temperature and pressure decrease. The proposed method provides for disabling the P-12 interlock and permits the use of additional condenser dump valves during cool down below 350°F which is the Mode 4 entry temperature. There is not an increase in the frequency of occurrence of an accident or malfunction of a component important to safety previously evaluated in the UFSAR. Overcooling events involving increased heat removal by the secondary system are analyzed in Chapter 15 of the UFSAR. These include "Excessive Heat Removal Due to Feedwater System Malfunctions" associated with a rapid increase in steam flow, "Excessive Load Increase Incident" associated with a rapid increase in steam flow, "Accidental Depressurization of the Main Steam System" associated with an inadvertent opening of a single condenser dump, relief or safety valve, and "Major Rupture of a Main Steam Line." Since the reactor is shutdown, Mode 4 shutdown margin will be assured by increasing boron concentration (i.e., Mode 4 at 200.1°F concentration) prior to allowing additional condenser dump valves overcooling event involving return to criticality is not credible at this phase of shutdown operation. The RHR system is available as before to prevent uncontrolled heat up of the RCS. The steam generator power operated relief valves are also available as a means of primary cooling via the secondary main steam system. Thus, the probability of an uncontrolled heat up due to failure of a cooling system or component (e.g., steam dumps) is not increased. The consequences of the accidents and malfunctions of the associated equipment remains unchanged whether the three cool down valves or the additional steam dump valves are used. The potential to create a new type of event not previously evaluated in the UFSAR is not found with this change. No new accidents are created by this change and no new control features have been incorporated. The disablement of the P-12 interlock will be performed procedurally with no permanent hardware modifications to the unit. The use of this cool down method does not affect the design basis limit of any fission product barriers. The rate of cool down is maintained within the acceptance limits

SUMMARY LISTING AND DESCRIPTION OF THE CHANGES
RESULTING IN A CHANGE TO THE CONTENT/VERBIAGE IN THE UNIT 1 UFSAR
CHAPTER 7 AS CONTAINED IN AMENDMENTS 0 THROUGH 8

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			for RCS cool down, as specified in the technical specifications (which ensure compliance with 10 CFR 50 Appendix G). The plant cool down is controlled in the same manner as previously considered which is manual operator action. The method of evaluation for the identified accidents is not modified nor is a new method created. This change does not have an impact on evaluation methodologies described in the UFSAR. Therefore, this change does not require a license amendment.
7	7.3	WBPLMN-07-014-0 D-52220-A 1917	DCN 52220-A modifies the Train A and Train B electrical circuits for high radiation in the refueling area logic bus and the Train A and Train B Solid State Protection System (SSPS) input for high radiation in the containment purge air exhaust, which initiates Containment Vent Isolation (CVI). The modification to these circuits allows WBN to maintain the Auxiliary Building Secondary Containment Enclosure (ABSCE) in the event of a high radiation signal in either the refueling area or the containment purge exhaust during refueling operations while the containment and/or annulus is open to the auxiliary building ABSCE spaces. This change will permit, but not require, operation of containment purge when moving irradiated fuel in the Auxiliary Building with containment and/or annulus hatches, personnel hatches, or penetrations open to the ABSCE spaces. This change does not allow the equipment hatch to be open during fuel movement inside containment since Technical Specification 3.9.4, "Refueling Operations, Containment Penetrations," requires the penetration to be closed during this time. The electrical circuit modifications are implemented in two Auxiliary Relay Racks. One hand switch for each train will be added to the appropriate rack to allow the circuits to be swapped from a normal mode to a refueling mode. During normal operation, the circuits will perform the same logic as before

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			the modification. Prior to starting refueling operations, the hand switch will be placed into the refueling mode, and it will be returned to the normal mode prior to entering Mode 4 from Mode 5. During the refueling mode, the ABSCE isolation valves will be closed by receiving either the current high radiation signal from the spent fuel pool accident radiation monitors or upon a CVI signal. The Containment Ventilation Isolation Valves that are currently closed by a CVI signal will also be closed upon receiving a high radiation signal from the spent fuel pool accident radiation monitors. Isolation of these valves will establish the ABSCE boundary, prevent the potential for back flow from the Shield Building exhaust vent, and ensure that radioactive releases due to fuel handling accidents in containment will be processed by the Auxiliary Building Gas Treatment System (ABGTS) if they migrate into the Auxiliary Building. The response time requirement for the spent fuel pool radiation monitors is not affected by this modification. A new response time requirement for the containment purge monitors during refueling operations is required. The new response time requirements do not affect the calculated off site or main control room dose analyses for the fuel handling accident. This change complies with the safety and functional requirements specified in the applicable design bases documents and does not adversely affect the performance of any safety related equipment. The proposed modifications do not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction beyond the ten percent allowed by 10CFR50.59, or create a new type of accident. The design bases for fission product barriers will not be altered or exceeded and no new

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			methods of evaluation were used in evaluating the proposed modifications.
0	7.5	WBPLEE-96-031-0 N/A 1418	This change, FSAR Change Package No. 1418, updates FSAR Section 6.3.5.4 and Table 7.5-2 (Sheet 5 of 18) to describe the Safety Injection System (SIS) Cold Leg Accumulator (CLA) Tank level measurement system. Specifically, Section 6.3.5.-4 states that two level channels are provided for each tank. The installed configuration uses one thermal dispersion type monitor (supplied by FCI) and one differential pressure transmitter (supplied by Rosemount). This change deletes the term "identical". Also, Table 7.5-2 (Sheet 5 of 18) identifies the CLA indication range to be 7632 to 8264 gallons. The design range of the CLA Tank level channels is 7450 to 8080 gallons. This change revises the FSAR to reflect the design range. Also, this change revises Design Criteria, WB-DC-30-7, "Post Accident Monitoring Instrumentation" to reflect the proper range. This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failure are created. Technical Specification is not affected. This change is in compliance With safety classification requirements as specified in design basis documents. The Safety Evaluation Report (SER) and Supplemental Safety Evaluation Reports (SSERs) Nos. 1 through 20 have been reviewed and no impacts were found. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exist.
1	7.5	WBPLEE-98-010-1 M-39608-A 1508	This DCN (M-39608-A) replaces the containment sump level transmitters in Unit 1. The existing transmitters are Barton transmitters with a diaphragm seal and capillary tubing. These existing transmitters have a problem with the capillary tubing leaking fill fluid, and maintaining the transmitter within calibration. The new transmitters are Class 1E qualified, do not have capillary tubing, are more accurate, and can be submersed during a LOCA. This change upgrades equipment used to perform a function. Functional performance of the plant is not affected and protective logic is not affected.

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			The range of the transmitter is changing from 0 to 20 feet (240") to 0 to 200" (16' & 8") which will improve instrument loop accuracy. The existing setpoint for switchover from RWST remains the same. The new transmitters' range is fully adequate to monitor the maximum equilibrium flood level, which is above the PAM requirement of 600,000 gallons. The sump is in the lower containment, below the refueling cavity. The sump is a water source for long term recirculation for the functions of RHR, emergency core cooling, containment atmosphere cleanup, and containment long term cooling. The transmitters will be located just outside the sump in the raceway. These transmitters are associated with the protective features used to detect and mitigate the effects of Condition III & IV events associated with a LOCA. Four safety-related level transmitters (one per channel) are provided to measure the containment sump level. These transmitters provide input to allow switchover from RWST to containment sump recirculation and also provide input to PAM Category I indicators 1-LI-63-180 and -181. The four containment sump level high trip signals are combined in a 2 out of 4 circuit to produce an output that is combined with the output of the RWST low level switches. When this logic signal is made up the valves from the RWST are closed, and the containment sump becomes the water source for long term recirculation. Implementation of this DCN requires the mounting of the new transmitters, rerouting instrument sense lines, and cables, and revising the dropping resistor at the Eagle racks. The setpoint will not change, and indication scales are not affected as they currently read in 0-100% scale. This change upgrades existing plant equipment. The failure modes of the replacement equipment do not differ from the equipment being replaced, and common mode failure has been demonstrated not to be an issue based on experience with these types of transmitters at Sequoyah. The installed loops (equipment and cable/conduit) are separated physically and electrically. Proper separation/isolation of cable routing and

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			equipment is maintained by the DCN and appropriate plant installation procedures. The independence of safety related equipment is not challenged. Civil calculations have been performed to verify that, when installed per the DCN, the equipment will remain able to perform its function following a seismic event. The equipment has been tested and the test report reviewed documenting the new transmitters are not to be susceptible to EMI/RFI and will not cause radiated emissions outside the requirements of the design standard and adversely affect the operation of surrounding equipment. This change will not compromise the ability of plant safety-related equipment to perform its intended function. Westinghouse calculation, WCAP-12096 shows the loop accuracy is within the previous calculated loop accuracy, therefore, with the swapover setpoint unchanged, the safety margin is not affected.
1	7.5	WBPLEE-98-093-0 WBPLEE-98-093-1 M-39911-A 1560	DCN M-39911-A, replaces the obsolete Unit 1 Westinghouse P2500 Plant Process Computer and consolidates the Emergency Response Facilities Data System (ERFDS) into a new Plant Integrated Computer System (ICS). The ICS provides an operator friendly, state of the art, real time process computer system for the WBN plant operators and Emergency Operation Facility (EOF) personnel. The new ICS computer is operational and performing all functions of the old P2500 computer and most of the functions of the ERFDS, including Safety Parameter Display System (SPDS), Bypass and Inoperable Status Indication (BISI), Balance of Plant (BOP), NSSS, Communications Data Links, and RHR Mid-Loop Operation Monitoring Functions. Essentially all "at power" design basis accidents are associated with the ICS because accident analysis assumes reactor conditions are within Technical Specification conditions. Several of these Technical Specification parameters are monitored with ICS computer software including thermal power, axial flux difference (AFD), Quadrant Power Tilt Ratio (QPTR), Rod Supervision, heatup/cooldown, and RCS inventory. The power range nuclear instrument gains are adjusted based on calorimetric calculations completed by the ICS. This ICS calorimetric calculation software will be designed, developed, and tested in accordance with TVA procedures. Test case results of the

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			calorimetric calculation and the other calculations identified above from the old P2500 are compared to similar calculations made by the ICS as part of the validation testing. The procedure requirements include formal test cases for ICS software as well as informal supplemental testing to further demonstrate software features and challenge calculation algorithms. Therefore, there is a high degree of confidence that the ICS Technical Specification compliance calculations are correct. The ICS is not safety related and is property isolated and separated from safety related equipment. It is designed to seismic Category 1(L)B criteria inside seismic Category 1 areas. In the event of an accident, Main Control Room (MCR), and Emergency Response Facility (ERF) personnel can use the SPDS and other aspects of the ICS as an aid to restore the plant to a safe condition. However, Operators must be trained to respond to accidents both with and without the SPDS available. The ICS was not designed to safety system criteria, and it is not used to perform functions essential to the health and safety of the public. Although the ICS indirectly provides support to safety related systems by alerting operators that an abnormal condition may exist, operators cannot procedurally take inappropriate safety-related action based solely on ICS information. There is other safety grade equipment that is provided for mitigating the events of design basis accidents.
2	7.7	50475 D-50475-A 1619	DCN D-50475-A replaces the existing Anticipated Transients Without Scram (ATWS) Mitigation System Action Circuitry (AMSAC) microprocessor based logic circuitry with relay-based logic circuitry. Modifications are within the existing AMSAC cabinet and only internal components that perform the AMSAC operational logic functions are within the scope of this DCN. AMSAC inputs (turbine first stage impulse pressure input signals from AMSAC dedicated transmitters and existing narrow range steam generator level transmitters) are not changed by this DCN. Similarly, AMSAC output interfaces, including safety-related isolation devices, with the auxiliary feedwater (AFW) system and turbine trip circuitry remain unchanged.

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			AMSAC has no primary safety functions. AMSAC performs secondary safety functions in that it is of sufficient quality and reliability to perform its intended function without contributing to transients which may challenge safety systems. AMSAC does not degrade the existing reactor protection system. The DBEs for the AMSAC are LONF/ATWS and the Loss of Load/ATWS. The scenarios of those two events are described below: A complete loss of normal feedwater occurs as a result of a malfunction in the feedwater/condensate system or its control system from such causes as the simultaneous trip of all condensate pumps, the simultaneous trip of all main feedwater pumps or the simultaneous closure of all main feedwater control, pump discharge or block valves. Because if a postulated common mode failure in the RPS, the reactor is incapable of being automatically tripped when any of several plant process variables have reached their reactor trip setpoints. The most severe plant conditions that could result from a loss of load occur following a turbine trip from full power when the turbine trip is caused by a loss of main condenser vacuum. Because of a common mode failure in the protection system, the reactor is incapable of being automatically tripped as a result of the turbine trip or as the result of any several other reactor trip signals that occur later in time when several plant process variables reach their reactor trip setpoints. The components installed to perform the AMSAC logic functions have been utilized in other nuclear applications to demonstrate a history of quality and reliability. The relay-based logic circuitry has been designed that output relays shall be energized to actuate in order to prevent spurious trips and false status indication on loss of power or logic. Therefore, no new failure modes are introduced by this modification.

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			The purpose of the AMSAC system remains the same: To function diverse from the reactor trip system to automatically start the AFW pumps and to initiate a turbine trip under conditions indicative of an ATWS. This modification does not change any system trip setpoints, and loop accuracy, as shown by calculation, is within the previous calculated loop accuracy, therefore, this modification will support the operational limits defined for AMSAC. For these reasons, this activity does not constitute a unreviewed safety question.
3	7.7	WBPLMN-01-084-0 D-51102-A 1702	This evaluation evaluates the effect of revising the control system logic to eliminate the actuation of the pressurizer backup heaters on a high level deviation signal. At present, a bistable is actuated on a high level deviation; the output of this bistable is used to both actuate an alarm on the control board and to actuate the backup heaters. The functional change is to eliminate the backup heater actuation on the bistable actuation, but to maintain the control board annunciator actuation. The pressurizer level control system is used to control the RCS water inventory whenever a steam bubble is present in the pressurizer. This system also provides indications, alarms and manual controls for operator monitoring and control. The pressurizer level control system utilizes three level channels to generate signals for pressurizer level control, anticipatory alarms, indication, and recording. The level channel inputs are isolated outputs from the reactor protection system. The three level channels provide individual indication on the MCR board. During power operation, the pressurizer level setpoint is varied as a linear function of Tavg as previously described. The difference between the level signal selected for control and the reference level is compensated to eliminate any steady-state error between indicated and its setpoint. The output signal is the automatic charging flow demand signal. The uncompensated level error signal feeds two bistables. One bistable is used to provide a low level deviation alarm. This alarm is set below the programmed

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			level by a preset amount to warn the operator that the charging system is not supplying enough charging flow to maintain pressurizer level. The second bistable actuates a high level deviation alarm if the measured level increases above the programmed level by a preset amount. This bistable also energizes the pressurizer backup heaters. The LONF/LOOP events are currently analyzed for the purpose of showing the AFW has sufficient long-term heat removal capacity. The single failure assumed for the LONF/LOOP event is failure of the turbine driven AFW pump which is the most limiting failure. Since the pressurizer backup heaters and their control circuits do not perform a primary safety function, single failure criteria does not apply. The definition of a single failure is a failure which results in the loss of a safety-related component to perform its intended safety function. Failures in systems not required to mitigate an accident are assumed when the component is in the zone of influence of the accident and is not designed and specified to remain functional in the accident environment. Since there is no zone of influence for this transient, an unintended spurious operation of the backup heaters is not required to be assumed. UFSAR Section 7.7 documents that the pressurizer water level control system is not required for safety. Analysis has shown that the plant design transients are not impacted by the change, and the safety analyses are not adversely impacted. In addition, implementation of the logic change does not make any changes to the protection grade isolated inputs into the pressurizer water level control system. Therefore, based on the above, the logic modification to eliminate the automatic backup heater actuation on a high pressurizer water level deviation, and maintain manual heater actuation capability, will not adversely affect the UFSAR described design functions of any structure, system, or component (SSC). Therefore, the control system logic modification to the automatic backup heater actuation on high pressurizer water level does not require a license amendment.

ENCLOSURE 2

TVA LETTER DATED SEPTEMBER 9, 2010
WATTS BAR NUCLEAR PLANT (WBN) UNIT 2
REQUEST FOR ADDITIONAL INFORMATION REGARDING FSAR
RELATED TO INSTRUMENTATION AND CONTROLS (TAC NO. ME2371)

SUMMARY LISTING AND DESCRIPTION OF THE OTHER
10 CFR 50.59 EVALUATIONS

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1	7.1	Safety Evaluation WBPLEE-98-088-0 50049 1552	FSAR Change Package 1552 documents changes resulting from a review of Section 7.1. Items 1-3, 5-8, 10-14, and 16-18 below are corrections of non-numerical typographical errors, administrative changes, or minor editorial changes that do not change the intent. As such, they are considered to be non-significant changes, and do not require safety evaluation. Therefore, these non-significant changes will not be further addressed. Item 1 includes those non-significant changes for which no explanation is needed. The remaining items (4, 9, 15) are addressed in this safety evaluation. Items 19 and 20 describe design basis document changes which are necessary to ensure consistency with existing design and licensing bases. These additional changes are being made by EDC E-50049 and are also addressed in this safety evaluation. 1. (Pages 7.1-2 through 7.1-7, 12, 13, 16-19; and Table 7.1-1 Sheets 1-6): These page contain minor changes which do not need explanation and, therefore, will not be addressed individually. Examples of these are addition of acronyms, correction of reference or figure numbers, addition of cross-references to other sections which address related topics, and editorial changes such as verb tense, word choice, grammatical corrections. 2. (Page 7.1-1): Change the identification of the ANS event classifications associated with normal, transient, and faulted conditions to be consistent with the definitions given in Chapter 15. 3. (Page 7.1-3): Revise the definition of hot shutdown to be consistent with the Technical Specification definition. 4. (Page 7.1-6 and Table 7.1 -1 Sheet 1): Delete Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment." This document

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			describes acceptable methods of complying with GDC 55 and 56. There is no reference to RG 1.11 in UFSAR Section 6.2.4 which describes compliance with GDC 55 and 56. Previous revisions to the FSAR, incorporated in Amendments 52 and 69, deleted from Section 6.2.4 statements indicating that the WBN design met the requirements of the RG. Therefore, this change to Section 7.1 is consistent with previous FSAR changes and will result in consistency between UFSAR sections. There are no commitments to RG 1.11 in WBN design criteria WB-DC-30-16 and WB-DC-40-34; therefore, no design basis changes are required. This issue was documented in WBPER980417. 5. (Page 7.1-10): Delete redundant information - the design bases for the Vital Control Power System are given in Section 8.3. 6. (Page 7.1-12): Delete information which is provided in another section. The parameters which initiate safety injection are listed in Table 7.3-1. 7. (Page 7.1-12, 19): Add ISA-DS-67.04 1982, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," as Reference 4. The methodology of both this document and the Westinghouse Setpoint Methodology for Protection Systems, WCAP-12096, which is Reference 6 of Section 7.1, were used to determine protection system setpoints. This change is consistent with Section 7.2.1.2.4, which references both of these documents as a basis for establishing setpoints. 8. (Page 7.1-13, 14): Delete redundant information - the design bases for the separation of cables and raceways of redundant circuits are given in Section 8.3. 9. (Page 7.1-17): Add the nameplate color code requirements for Post-Accident

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			Monitoring equipment located inside the main control room (MCR). The listed color codes apply only to components located outside the MCR. This addition is consistent with the existing design basis and will not require any changes in the identification of PAM equipment in the plant. 10. (Page 7.1-19; Table 7.1-1 Sheets 3 and 5): Delete Reference 4 (WCAP 10271) and Reference 7 (WCAP 7486) since these documents are not referenced in the text of Section 7.1 and, therefore, should not be listed to conform to WBN FSAR convention. Both of these documents are, however, referenced in Notes 1 and 3 of Table 7.1-1. These notes are modified to fully identify the reference documents within the notes since there is not a list of references in the table. 11. (Table 7.1-1 Sheet 1): Delete Regulatory Guide 1.40, "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants," and Regulatory Guide 1.73, "Qualification Tests for Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants." These documents are not applicable to the plant instrumentation and are also addressed in Section 8.1.5.3, which indicates full compliance with both of these documents. 12. (Table 7.1-1 Sheet 1): The table indicates full compliance with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." It also refers to Note 7 of the table which then refers to Section 5.2.7 for discussion of compliance. This change eliminates the full compliance notation from the table so that compliance is discussed in only one place. Section 5.2.7 states that the leakage detection systems comply with applicable parts of GDC 30 and RG 1.45. 13. (Table 7.1-1 Sheet 2): The table indicates full compliance with IEEE Standard

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			308-1971, "Class 1E Power Systems for Nuclear Power Generating Stations." Compliance with this document for the electrical systems which provide power to the safety related plant instrumentation is discussed in Chapter 8. Section 8.1.5.3 indicates that the WBN electric power system design meets the intent of the standard. This change eliminates the full compliance notation from the table and adds a reference to Chapter 8 so that compliance is discussed in only one place. 14. (Table 7.1-1 Sheet 2): Add IEEE Std. 323-1971, "IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and IEEE Std. 379-1972, "IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems," and reference appropriate notes. Conformance to these standards is already provided in the table notes. IEEE 323-1971 is referenced in the discussion of compliance with Regulatory Guide 1.89 in Note 4 of the table. Compliance with IEEE 379-1972 is discussed in Note 3 of the table. 15. (Table 7. 1 -1 Sheet 3): Note 1 of the table discusses conformance to the periodic testing requirements of IEEE Std. 338-1971. Item 2 of the note discusses development of reliability goals and adequacy of test frequencies but does not relate the two. Although specific goals were not developed for protection system reliability, the evaluation of test intervals in WCAP-10271 Supplement 1 and WCAP-1 0271 -P-A Supplement 2, "Westinghouse Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrument System," established that the test frequencies are adequate to confirm acceptable protection system reliability consistent with risk assessment results.

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			16. (Table 7.1-1 Sheet 4): Note 2 of the table lists some of the equipment and functions which are not tested during operation because of the risk of plant upset and describes the basis for not testing such equipment at power. The intent of this list was to identify examples of plant equipment which is not tested at power; it was not intended that this be a complete list. This change provides clarification. 17. (Table 7.1-1 Sheet 5): This change clarifies that the test circuitry being discussed is part of the SSPS. 18. (Figure 7.1-2): Sheet 1 of the figure is drawing 45W1640. This drawing was originally one sheet but was expanded to two sheets to include additional design information. This change adds 1-45W1640-1 as sheet 2 of the figure and adds sheet 1 to the existing figure number. The drawing will added to the UFSAR per NADP-7. In addition to the UFSAR changes described above, design basis documents (DBD) are revised by EDC E-50049 to clarify the design basis and functions, correct minor errors, and make editorial changes to maintain consistency between the DBD and the UFSAR. Specifically, System Descriptions N3-38-4002, Auxiliary Feedwater, and N3-99-4003, Reactor Protection, are revised as follows and there are no associated UFSAR changes: 19. UFSAR Section 7.1.2.2 states that exceptions to instrument sense line independence requirements will be documented in design basis documents. System Description N3-36-4002 does not document an exception for the use of common sense lines for Auxiliary Feedwater flow transmitters for steam generator loops 2 and 3 (1 -FT-3-155A and B, 1 -FT-3-147A and B). This problem is identified in WBPER980417 and is resolved by addition of an exception to N3-3B-4002. The basis for this exception was previously documented in N3E-934 by DCN P-03131 -A. This change is documentation

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			only and no UFSAR changes are required. 20. Section 2.2.8 of N3-99-4003 is revised by EDC E-50049 to clarify the requirements for separation of redundant protection system channels (1, 11, 111, and IV) and separation of the four protection set channels from the two logic trains (A and B). The change also clarifies that the requirements apply to the Essential Safety Features Actuation System (ESFAS) as well as to the Reactor Trip System. The changes are consistent with UFSAR Section 7.1.2.2.2. These changes do not involve any physical modifications to the plant or modify the safety function of any equipment. The changes do not alter any design basis accident or operational transient analyses previously performed, and no new accidents or equipment malfunction failures are created. The changes do not affect setpoints or safety limits and, thus, do not reduce any margins of safety as defined in the Technical Specifications. Therefore, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question.
1	7.2	Safety Evaluation WPBLEE-98-080-0 50045 1553	FSAR change package 1553 documents changes resulting from a review of UFSAR Section 7.2. Items 1-15 and 17-38 are corrections of non-numerical typographical errors, administrative changes, or minor editorial changes that do not change the intent. As such, they are considered to be non-significant changes as defined in Nuclear Assurance Department Procedure (NADP) 7, "FSAR Management," Section 5.0, and do not require an SA, SR or SE. Therefore, these non-significant changes will not be further addressed. Item 1 includes those non-significant changes for which no explanation is needed. The remaining items (16 and 39) are addressed in the Screening Review and Safety Evaluation. Design basis document changes are being made by EDC E-50045 to ensure consistency with existing design and licensing bases, including changes associated with the UFSAR changes described below.

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			16 (page 7.2-14, 34, 35): The update to Section 7.2.1.1.5 is taken from text in Section 7.2.2.3.4 with clarifications and editorial changes. The relocated discussion of the pressurizer water level instrumentation is more appropriately included in this section than Section 7.2.2.3.4, which deals with control and protection system interaction. The changes to 7.2.1.1.5 are based on a general description of the Westinghouse pressurizer level design, channel independence, and actual installation attributes found on TVA physical drawings. Also, the hydrogen gas entrainment issue documented in NRC Information Bulletin Number 92-54, Level Instrumentation Inaccuracies Caused by Rapid Depressurization, is retained and clarified. Similar clarification is made to Reactor Protection System Description N3-99-4003 Section 3.1.1.2(d). The original text in 7.2.2.3.4 provides some information that is too detailed and is not pertinent to the subject of discussion. It also includes a statement that the error effect on the level measurement during a blowdown accident would be about one inch. The basis for this value is not known; however, the worst case reference leg loss of fill error due to a rapid RCS depressurization event is no more than 12 inches elevation head. This value is based on the relative elevation difference between the condensing chamber and the reference leg sensor bellows. The channel error value discrepancy is documented in a WBN PER. The remaining text in 7.2.2.3.4 is revised to clarify the control and protection system interaction discussion. 39 (Figure 7.2-1 Sheet 3): This drawing, 1-47W611-99-6, shows time delays of 0.5 and 0.1 seconds, respectively, for Reactor Coolant Pump undervoltage (UV) and underfrequency (UF) reactor trip signals. Setpoint and Scaling Documents specify settings of 23 cycles (0,383 sec) for the UV and 5 cycles (0.087 sec) for the UF as determined by calculations WBPE0689009007 and WBPE0689009008. This discrepancy is documented in WBPER980417. The

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			drawing will be revised by DCN E-50045. In addition to the UFSAR changes described above, Reactor Protection System Description N3-99-4003 is revised by EDC E-50045 to clarify the design basis and functions, correct minor errors, and make editorial changes to maintain consistency between the design basis and the UFSAR. Also reference 7.5.26 of N3-99-4003 is changed from WCAP-14419 to WCAP-14738, "Revised Thermal Design Procedure Instrument Uncertainty Methodology," to reflect the current design and licensing basis for the RCS flow and reactor power calorimetrics instrument uncertainty, which became effective at cycle 2 startup. WCAP-14419 was superseded by WCAP-14738. These documents are not listed in the UFSAR or TS. These changes do not involve any physical modifications to the plant or modify the safety function of any equipment. The changes do not alter any design basis accident or operational transient analyses previously performed, and no new accidents or equipment failure modes are created. The changes do not affect setpoints or safety limits and, thus, do not reduce any margins of safety as defined in the Technical Specifications. Therefore, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.
1	7.3	Safety Evaluation WBPLEE-98-089-0 50048 1554	FSAR change package 1554 documents changes resulting from a review of Section 7.3. Non-significant changes items 1-25 and 27 will not be further addressed. Item 1 includes those non-significant changes for which no explanation is needed. The remaining item (26) is addressed in this safety evaluation. Item 28 describes design basis document changes which are necessary to ensure consistency with existing design and licensing bases. This additional change is being made by EDC E-50048 and is also addressed in this safety evaluation. 1. (Pages 7.3-1-8, 11, 16, 17, 20; and Table 7.3-3 Sheet 1): These pages contain minor changes which do not need explanation and, therefore, will not be

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			addressed individually. Examples of these are addition of acronyms, correction of reference or figure numbers, addition of cross-references to other sections which address related topics, and editorial changes such as verb tense, word choice, grammatical corrections. 2. (Page 7.3-2, 5): In the list of functions which are initiated by the Engineered Safety Features Actuation System (ESFAS) (Section 7.3.1.1.1), combine and simplify items 2 and 3, both of which describe Emergency Core Cooling System (ECCS) functions. Remaining items are renumbered. Similarly in Section 7.3.1.1.4, change safety injection to ECCS. ECCS is a more broadly descriptive term which includes safety injection. System Description N3-99-4003 is similarly revised. 3. (Page 7.3-3, 5): Also in the list of ESFAS-initiated functions in Section 7.3.1.1.1, revise and simplify Item 4, Auxiliary Feedwater (AFW), and include AFW valves since these values are also actuated by ESFAS. Similarly, add the AFW valve actuators to the list in Section 7.3.1.1.4. 4. (Page 7.3-4): Delete orifice plates from the list of device types used in the measurement of protection system variables. Orifice plates are a subset of flow elements, which are also listed. In addition, orifice plates are not used as sensors for protection system variables. 5. (Page 7.3-4): Revise Item 3 of Section 7.3.1.1.2 to simplify the discussion of valve position information available during the post-LOCA recovery period. 6. (Page 7.3-5): Clarify that, in addition to the safety injection lines, the containment spray lines also are not isolated by a Phase B containment isolation signal. This change is consistent with Section 7.3.1.1.1.

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			7. (Page 7.3-5): Add the Auxiliary Building Gas Treatment System, Emergency Gas Treatment System, and Motor-Operated Valve Thermal Overload Bypass to the list of equipment actuated by the ESFAS. This change is consistent with the discussions of these features in the referenced chapters of the FSAR. 8. (Page 7.3-6): Revise the section summarizing the generating station conditions which require protective action. The list is not intended to be a complete list of the design basis events which the protection system is designed to mitigate. The change simplifies the summary, adds feedwater line break, and adds a reference to Chapter 15 for identification of the conditions requiring protective action. System Description N3-99-4003 is similarly revised. 9. (Page 7.3-7 and Table 7.3-2, Item 3): Revise the summary of the generating station variables which are required for initiation of protective action by the ESFAS. The change simplifies the summary, eliminates repetition, and adds steam generator level and reactor coolant temperature (Tavg) as monitored variables. Low-low Steam Generator (SG) level starts AFW. High-high SG level initiates feedwater isolation. Low Tavg coincident with a reactor trip also initiates feedwater isolation. Low Tavg, with a note to identify the interlock with Permissive P-4 (reactor trip), is also added to Table 7.3-2, item 3, which lists the conditions that initiate Feedwater Isolation. Addition of these variables is consistent with discussions of the Main and Auxiliary Feedwater Systems in Sections 10.4.7, 10.4.9, various Chapter 15 events (e.g., Sections 15.2.10, 15.3.1, 15.4.2), and Technical Specification Bases 3.3.2 for the P-4 interlock. System Description N3-99-4003 is similarly revised to add SG level and reactor coolant temperature. 10. (Page 7.3-8): This change is a clarification, replacing loss of coolant and

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			steamline break with a more general term, design basis events, which also includes feedwater line breaks. A reference to Chapter 15 is also added for identification of the postulated events for which the ESFAS is required to actuate. 11. (Page 7.3-8): Revise the list of typical ranges of the instrumentation required for initiation of protective action by the ESFAS. The change simplifies the summary; eliminates repetition; replaces the terms loss of coolant and steamline break with a more general term, design basis events; and adds a reference to Chapter 15. SG level and Tavg are added to the list since these variables actuate ESFAS as described in Item 9 above. System Description N3-99-4003 is similarly revised. 12. (Page 7.3-9): Editorial change to more accurately describe the drawings which reflect the design of the protection systems. 13. (Page 7.3-9): Revise the discussion of the failure mode and effects analysis performed for the ESFAS. This change simplifies the discussion and eliminates unnecessary detail. The reference provided in the section describes the analysis in detail. 14. (Pages 7.3-10, 12, 13): As with reactor trip channels, most ESFAS channels are designed so that loss of instrument power results in trip of the ESFAS channel, i.e., the protection system comparator output is normally energized and de-energizes to actuate. Containment spray is identified as an exception to the typical design in order to avoid spurious actuations. In addition to the containment spray function, the switchover from injection to recirculation following a safety injection is also designed so that the comparator output energizes to actuate. This change adds the switchover function as an exception consistent with the switchover discussion in Section 7.6.9.5. System Description N3-99-4003 is similarly revised.

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			15. (Pages 7.3-10): This change provides clarification of the manual controls provided for containment spray actuation. The discussion notes that there are two sets of switches with one set/train and two switches/set. The change is to clarify that simultaneous operation of both switches in either set will actuate containment spray in both trains (i.e., the sets are not aligned with a specific train). This is shown on Figure 9.4-30 and is described in Technical Specification Bases 3.3.2 for containment spray. 16. (Page 7.3-13): The discussion of online testing of the ESFAS and ESF actuators notes that there are exceptions to the normal test procedure which results in operation of the ESF device. This change clarifies that the exceptions are for devices which cannot be operated at power without causing plant upset. This is consistent with the more detailed discussion of an exception for such equipment in Table 7.1-1, which is referenced in this section. 17. (Page 7.3-15): Clarify that the ECCS and containment spray system tests are performed as described in the applicable sections of Chapter 6 and in accordance with the Technical Specifications. This is consistent with discussions of testing in Sections 6.2 and 6.3. 18. (Page 7.3-16): Delete momentary from the description of the main steam isolation valve control switches. The switches are rotary type with spring return, i.e., momentary, from the OPEN position; the CLOSE position is maintained. This is consistent with Figure 10.3-5 (drawing 1-47W611-1-1). 19. (Page 7.3-17): Change preferred operating position to preferred failure position in the description of the position pneumatically operated valves assume upon loss of control air. Failure mode or position is a more commonly used term when

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			describing the response of components to loss of motive power. 20. (Page 7.3-17, 18 and Table 7.3-1): The initiating signals for AFW are moved from Section 7.3.2.3 to Table 7.3-1, which lists ESF instrumentation. A reference to the Table is added. This change also clarifies that the AFW pumps are started by trip of both Turbine-Driven Main Feedwater (MFW) pumps rather than all MFW pumps as currently stated since trip of the Standby MFW pump does not initiate AFW. This is consistent with the description of the AFW System in Section 10.4.9. This change also deletes ATWS Mitigation System Actuation Circuitry (AMSAC) from the list of AFW start signals. As described in Section 7.7.1.12, the AMSAC system is non-safety and provides a diverse means of initiating AFW and turbine trip under conditions indicative of an ATWS event. AMSAC was not designed as an Engineered Safety Feature and is not included in the ESFAS Technical Specification 3.3.2 for AFW start. Therefore, it does not belong in the Table which identifies ESF instrumentation. The change does not after the AMSAC functions of AFW start and turbine trip. The switchover from injection to recirculation and the switchover initiating signals are also added to Table 7.3-1 since they are considered to be part of the ESFAS. The listing of switchover instrumentation is consistent with the description of the switchover function in Section 7.6.9. Also numbered the notes at the bottom of the table. 21. (Page 7.3-19): This section specifies that out of service channels are placed in the trip mode except containment spray channels, which are placed in the bypass mode. Other channels are also placed in bypass when out of service, e.g., RWST and containment sump level channels, which initiate switchover from injection to recirculation. Since the Technical Specifications dictate the required mode for an out of service channel, the exception for containment spray is replaced with a reference to the Technical Specifications.

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			22. (Page 7.3-20): Delete the response time for generation of the protection system signal for steamline break protection since it is given earlier in the same section. Also delete the closing time of the main steam isolation valves. This information is provided in Section 10.3 and is therefore redundant. 23. (Table 7.3-2, Item 1): In the list of containment isolation (CI) initiators, delete "automatic" from "automatic safety injection" since manual SI also actuates CI Phase A as shown on Figure 7.3-3 Sheet 4. 24. (Table 7.3-2, Item 2): Clarify that the high steamline pressure rate which initiates steamline isolation is a negative rate, consistent with Table 7.3-3 (P-1 1) and Sections 15.2.13 and 15.4.2. 25. (Table 7.3-2, Item 4b): Clarify that the containment gas monitor which initiates containment vent isolation (CVI) monitors the containment purge air exhaust. Also clarify that there are a total of two channels (one per train); only one is required for actuation. These changes are consistent with Sections 9.4.6 and 11.4.2.2.6, the CVI logic shown on Figure 7.3-3 Sheet 4, design basis document N3-30RB-4002, and Technical Specification 3.3.6. 26. (Table 7.3-2, Items 4c and 4d): Auxiliary Building gas and air particulate monitor high radioactivity do not initiate CVI as indicated in this table and, therefore, are deleted from the table. The CVI signal is provided to isolate the containment purge lines on detection of high radiation in the purge exhaust lines or in the event isolation of containment is otherwise required. This is accomplished by initiating CVI on high radiation from the purge exhaust monitors or on safety injection. The Auxiliary Building gas and air particulate monitors do not initiate CVI on high radioactivity as indicated in UFSAR Table 7.3-2. Deletion of these functions from the Table is consistent with UFSAR Sections 9.4.6 and 11.4.2.2.6, the CVI logic

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			shown on UFSAR Figure 7.3-3 Sheet 4, design basis documents, and Technical Specification 3.3.6. The change does not require any plant modifications and is consistent with the existing design and licensing bases for containment isolation. 27. (Table 7.3-3): Clarify that when Permissive P-4 is present (reactor tripped), automatic reactuation of SI can be manually blocked (i.e., after SI has been initiated and the reactor tripped). The present wording implies that with P-4 present, SI can be manually blocked/reset before automatic actuation of SI has occurred. This is not the case as shown on Figure 7.3-3, Sheet 3. Similarly, reactuation of SI (after SI reset) cannot be blocked when the reactor is not tripped. In addition to the UFSAR changes described above, Reactor Protection System Description N3-99-4003 is revised by EDC E-50048 to clarify the design basis and functions, correct minor errors, and make editorial changes to maintain consistency between the design basis and the UFSAR. Also the following problem is resolved by a revision to the Auxiliary Feedwater System Description N3-3B-4002: 28. The AFW system is required to start when both Turbine-Driven Main Feedwater Pumps (MFWP) trip. System Description N3-313-4002, Section 2.2.6.1 requires this feature to meet single failure criteria and be implemented with Class 1E circuits using redundant, coincident logic. This signal is derived from a single non-safety-grade switch on each pump and, therefore, does not satisfy the design basis requirement. N3-3B-4002 is revised to identify the MFWP trip signal as an exception to these requirements. This is acceptable based on the following: The loss of feedwater indicated by the MFWPs trip would be followed by a drop in SG levels which, on reaching the low-low level setpoint, would initiate reactor trip and AFW actuation. Thus, this signal provides earlier AFW start (and decay heat removal) than would otherwise be the case. Since AFW start by this signal is not credited in any safety analysis (i.e., not a required accident mitigation function), it

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			is not required that protection system criteria be applied to the components which initiate the signal. The interface of this signal with the Class 1E AFW circuits is accomplished in accordance with the applicable requirements of the WBN separation criteria. The Technical Specification for AFW start (3.3.2) requires only two channels and the Bases for the function note that each MFWP is provided with one pressure switch. These changes do not involve any physical modifications to the plant or modify the safety function of any equipment. The changes do not alter any design basis accident or transient analyses previously performed, and no new accidents or equipment failure modes are created. The changes do not affect setpoints or safety limits and, thus, do not reduce any margins of safety as defined in the Technical Specifications. Therefore, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.
1	7.5	Safety Evaluation WBPLEE-98-060-00 39931 1541	This DCN (M-39931-A) will replace the Emergency Response Facilities Data System (ERFDS) workstations in the Unit 1 Main Control Room (MCR) as well as the ERFDS workstations in the Technical Support Center (TSC). A new workstation will also be added in the TSC. The workstations in the MCR consist of a new Personal Computer (PC), which includes a central processing unit (CPU) and a monitor. The workstations in the TSC will consist of a new PC, including a CPU, monitor, and a mouse, The existing keyboards will remain since they are specially designed to support the existing ERFDS software which will be reinstalled on the new workstations. These workstations will be upgraded to the fastest CPUs currently available with new larger touch screen monitors (Liquid Crystal Displays LCDs) in the MCR and larger non-touch screen monitors (Cathode-Ray Tubes CRTs) in the TSC. The communications hub for the TSC workstations will be replaced with dual hubs by this modification and a new fiber optic (F/0) jumper will be added in the computer room to supply the signal to the second hub added in the TSC. All UNIDs for the ERFDS monitors will also be changed from "CRT" to "MON". The "MON" description

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			is more general and will allow any type monitor to be installed (CRT, LCD, etc..) without the need for UNID changes as technical advancements occur. The existing ERFDS line printer in the TSC will also be replaced with a new laser printer as part of this modification. ERFDS EMS acquires, processes, and displays all data to Support the assessment capabilities of the MCR, Technical Support Center (TSC) and the Emergency Operation Facility (EOF).The ERFDS also provides the safety parameter display system (SPDS) and the bypassed and inoperable status indications (BISI) system for WBN. ERFDS is not defined as being primary safety-related and it is not required to meet the single failure criterion or be qualified to IEEE criteria for Class 1E equipment. SPDS The principal purpose and function of the Safety Parameter Display System (SPDS) is to aid control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing if abnormal conditions require corrective action by the operators to avoid a degraded core. During emergencies the SPDS serves as an aid to evaluating the current safety status of the plant executing function-oriented emergency procedures, and monitoring the impact of engineered safeguards or mitigation activities. The SPDS also operates during normal operations, continuously displaying information from which the plant safety status can be readily and reliably accessed. The SPDS is not class 1E qualified and is not powered from a class 1E power source. As such, the SPDS is electrically isolated from equipment and sensors used in safety systems. The SPDS equipment must be installed so that it does not degrade existing safety systems. The SPDS is not a safety system but may result in an improvement to

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			safety. Operators must be trained to respond to accidents both with and without the SPDS available. The SPDS shall be designed to provide reliable indication during all modes of plant operation, although it is not required to withstand a design basis event. BISI The BISI system is a computer based system that provides automatic indication and annunciation of the abnormal status of each ESFAS actuated component of each redundant portion of a system that performs a safety-related function. The determination of the bypassed or inoperable status of a system is left up to the reactor operator. The BISI system does not perform functions essential to safety. No operator action is required based solely on the abnormal status indication. The BISI system has no effect on plant safety systems. This modification does not change the function of any of the systems described above. It only enhances the speed of the CPUs and size and type (CRT to LCD) of the monitors for the ERFDS workstations located in the MCR and TSC. The above systems are non-safety related and are properly isolated and separated from safety related equipment They not required to meet the single failure criterion or to be qualified to IEEE criteria for Class 1E equipment. There are no now single failures or equipment failure modes introduced by this modification. There are no analyzed design basis accidents (DBA's) directly associated with the ERFDS workstations. However, ERFDS is designed to provide a complete data set to permit accurate assessment of the event without interfering with emergency operations in the MCR. Upgrading the ERFDS CPUs and monitors will not adversely impact any previously completed analysis of DBA's. The replacement of the ERFDS state-of-the-art workstations does not create any new accidents of any type that would represent an unreviewed safety question.

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1	7.6	Safety Evaluation WBPLMN-98-117-0 50059 1584	This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification. Specifically addressed are UFSAR Sections 4.2.2, 4.2.3, 4.2.4, 6.2.1, 6.2.2, 6.2.4, 6.2.6 and 7.6.6 and the associated system descriptions and calculation changes which were identified as a result of that review. Regulatory Guide 1.70, November 1978, "Standard Format and Content of Safety Evaluation Reports for Nuclear Power Plants" was utilized in this verification and content effort. Any discrepancies between the UFSAR and the design documents such as system descriptions, design criteria, calculations, and drawings were investigated and documents revised as appropriate by EDC E-50059-A. These changes are for "documentation" only with no impact on WBN's design bases or operational configuration. The document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. The proposed "documentation only" changes to the UFSAR and the design documents will not increase the likelihood of the design basis accidents occurring. These changes to the UFSAR and the design documents do not effect physical changes to the plant. These documentation-only changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. The proposed changes do not adversely affect the plant design bases nor do they cause any changes to the physical plant. The credible failure modes for the systems affected by these changes, have been evaluated against the accidents identified in the FSAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the FSAR accidents. The applicable accidents and the equipment served by the affected safety systems have been reviewed against these documentation changes, and no new malfunction pathways will be introduced which have not previously been evaluated and identified. The bases of the Technical Specifications have been reviewed for determining if any margins of safety are affected by these documentation changes. No margin of safety

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			is identified in the bases section of the Technical Specifications which could be reduced by these changes. The change to FSAR section 7.6.6 to describe the removal of power to FCV-63-8 and -11 during residual heat removal (RHR) cooldown does not involve an unreviewed safety question. Removal of power in itself does not have the potential to increase the probability of any accidents previously evaluated in the FSAR. The change does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated. Power removal to these valves is only applicable during Mode 4. Power can be restored and the valves opened in the event that they are needed for long term containment sump recirculation in the event of a Mode 4 LOCA. The consequences of an accident previously evaluated are not increased. In the event of a Mode 4 LOCA, FCV-63-8 and -11 are opened to transition from the refueling water storage tank (RWST) injection mode to the containment sump recirculation mode. Previous analyses of required flow during a Mode 4 LOCA have determined that the flow of one centrifugal charging pump is adequate to maintain core cooling. The duration of the injection mode for a Mode 4 LOCA, with only a centrifugal charging pump (CCP) drawing suction from the RWST, is approximately 10 hours. The relatively low flowrate out of the RWST would allow time for Operations to evaluate restraints and restore power to the valves to accomplish the RWST to containment sump swapover. Therefore, safety injection/core cooling capability is not impacted by the change to FSAR section 7.6.6. The consequences of a malfunction of equipment important to safety are not increased. The design redundancy that exists by having both 1-FCV-63-8 and -11 for the recirculation mode is not impacted. Removal of power to FCV-63-8 and -11 during RHR cooldown does not create the possibility for an accident of a different type and does not create the possibility for a malfunction of a different type. Power removal during Mode 4 actually helps to prevent the over pressurization of the safety injection pump and CCP suction piping. Technical Specifications 3.5 for the ECCS system have been reviewed and the change to FSAR section 7.7.6 to

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			describe removal of power to FCV-63-8 and -11 during RHR cooldown does not have the potential to reduce any margins of safety defined in the bases for these Technical Specifications. Therefore, based on the above justifications, the proposed changes do not involve an unreviewed safety question and are acceptable from a nuclear safety standpoint.
1	7.6	Safety Evaluation WBPLEE-98-075-00 50023 1547	FSAR Section 7.8 is revised to reflect changes resulting from a complete review to the section. A copy of the entire section with reference number identified in included as Attachment A. Reference numbers are added to each change which correspond to the description numbers shown below. Items 1, 2, 3, 4, 5, 8, 7, 9, 10, 11, 13, 14, 15, 18, and 17 are considered editorial/clarification changes that do not change the intent of the text. The remaining items, 8 and 12, are not considered minor changes and are discussed further. Item 12 requires the issuance of Engineering Document Change (EDC) Number E-50023-A to correct Design Basis Document (DBD) Number WB-DC-30-31. 1. (FSAR Pages 7.8-1 & 7.6-2). - The RHR isolation valves interface with the RCS system is discussed in Section 7.6.2. An RCS high pressure interlock is used to prohibit opening the RHR isolation valves to prevent the over pressurization of the RHR system. The specific interlock value and associated MCR RCS high pressure alarm setpoint value is removed from this section and replaced with a functional description (i.e., RHR System design pressure limit) of these setpoint values. These specific setpoint values represent information that is considered too detailed and does not contribute to the understanding of the operation of the subject RHR isolation valves. This change does not affect any functional or operational features of the subject RHR isolation valves. The RHR isolation valve logic is shown an UFSAR Figure Number 7.6-6 Sheet 3. 2. (Page 7.6-1). - The term 'RCS' is added to the first and second paragraphs for

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			clarification purposes. This change does not affect the content of the discussion of the RHR isolation valves as given in Section 7.6.2. 3. (Page 7.6-1). - Corrected the Figure Number associated with logic drawing 1-47W611-74-1 as referenced in the fifth paragraph. The correct Figure Number is 7.6-6, Sheet 3, Figure Nos. 7.6-7 Sheets 1 & 2, are deleted. The logic sketches depicted are a simplified version of the logic information shown on Figure Number 7.6-6. Therefore, Figures Nos. 7.6-7, Sheets 1 & 2 are deleted to avoid a possible misinterpretation of the logic information shown. 4. (Page 7.6-1). - The last sentence in the fifth paragraph discusses the RHR isolation bypass valves. The term "letdown" is deleted for text consistency. This term is not used elsewhere in this Section 7.6.2 or in other references associated with the RHR isolation bypass valves. 5. (Page 7.6-2). - In the last sentence of Section 7.6.2, a reference is made to UFSAR Section 3.11 related to environmental qualification of the RHR isolation valves, This sentence implies that the environmental qualification of the subject valves are specifically discussed. Section 3.11 discusses the environmental qualification program of which the subject valves are a part. This change makes this clarification. This change does not affect any technical issues associated with the environmental qualification of the subject valves. 6. (Page 7.6-3). - The interlock/permissive logic for the SIS accumulator isolation valves is discussed. The valves receive an automatic open signal when the RCS pressure reaches the P-11 permissive setpoint. This permissive signal is currently described as the "safety injection unblock pressure" which is not consistent with other sections of the UFSAR (i.e., Table 7.3-3). This change corrects this consistency problem. This change does not affect any functionally or

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			operational features of the SIS accumulation valves. 7. (Page 7.6-4). - Section 7.6.6 discusses the potential spurious operation of various control valves. To preclude spurious valve operation, open and/or closed contacts were placed before and after the opening/closing coil, as required. The term *'redundant' is* used to describe the use of the above described contacts. The term redundant is used in the nuclear industry to describe two or more totally independent features. This term is changed to *'separate'* to more clearly describe this design feature. This change is considered to be a clarification and does not affect any functional or operational feature of the subject motor operated valves. 8. (Page 7.6-4). - Section 7.6.6, third paragraph, discusses the use of protective covers installed over MCR handswitch as for specified motor operated valves. This paragraph did not identify the handswitches associated with valves 1-FCV-62-98 and -99 as an exception to the use of these protective covers. This change corrects this incomplete statement. 9. (Page 7.6-4). - Section 7.6.8, third paragraph, second sentence, discusses the use of protective covers. This sentence was revised to change the verb from future to present tense. This is an editorial change and does not affect the discussion of the protective hand switch covers used in the main control room. 10. (Page 7.6-4). - Section 7.6.6, fourth paragraph, discusses the removal of motive power from specific motor operated valves during normal operation. The term *"normal operation" is* changed to *"specific modes of plant operation."* This change is considered a clarification since the term normal operation may imply full power operation. The power removal may occur during startup activities as directed by Technical Specifications. This change does not affect any functional or operational feature of the subject motor-operated valves.

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			11. (Page 7.6-5). - Section 7.6.7 describes the loose parts monitoring system (LPMS). The first paragraph discusses the sensor locations and physical separation installation. Three sentences provide detailed information of sensor location such as; sensors are stud mounted on the vessel head lifting lugs, etc. This information is too detailed and does not significantly contribute to the understanding of the LPMS installed at Watts Bar. These three sentences are deleted. This change does not affect LPMS performance as described in this UFSAR section. Also, the general sensor location described in this paragraph provides adequate information related to sensor location. 12. (Page 7.6-5). - Section 7.6.7, second paragraph, discusses LPMS functional features related to Regulatory Guide (RG) 1.133 requirements. Specifically, the computer-based analytical system is described to use statistical and spectral analysis in order to demonstrate channel performance. The statistical analysis feature of this system have not provided any meaningful results. The software program that performs this statistical analysis was found to be difficult to administer without consultation with the (vendor supplied) software program's originator. Also, TVA engineering staff determined the program to be of limited benefit. Therefore, this feature is deleted from this section. The spectral analysis feature has proven to be a valuable analysis tool. This feature provides a frequency versus amplitude plot of any selected channel and provides a spectral signature of acoustic energy generated by primary loop equipment during normal plant operation. Normal plant operating noise is dynamic which provides an excellent basis for sensor operability determination. This spectral data is compared to previous data taken for each sensor in order to trend sensor operating characteristics for possible sensor degradation (i.e., changes to bandwidth/amplitude data) or complete sensor failure. Channels which exhibit repeatable spectral data is considered to be operational and capable of accurately

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			converting acoustical energy to electrical signals for processing by the LPMS signal conditioning circuitry. The recording and evaluation of this spectral data is considered sufficient to meet commitments discussed in this section of the UFSAR related to channel performance and channel calibration requirements. Since this change reflects a deviation to commitments made in this section of the UFSAR and the SER (Supplement Number 16), this condition is documented in the PER. 13. (Page 7.6 -5). - Section 7.6.7, third paragraph, last sentence, describes the LPMS's background noise averaging feature. This feature measures the background noise signal and adjusts the impact alarm monitoring circuitry to detect acoustic energy that occurs above this background noise. The term "maximum" is used to describe this automatic sensitivity adjustment The term is replace by *'high'* to more precisely describe this feature. This change does not affect the function or operation of the LPMS as described in the section. 14. (Page 7.6-6). - Section 7.6.7, second paragraph, discusses sensor location for the secondary side monitoring. The term *"trunnion"* is deleted associated with the Steam Generator sensor installation. This information is too detailed and does not significantly contribute to the understanding of the LPMS installed at Watts Bar. The general sensor location described in this paragraph provides adequate information related to sensor location. This change does not affect LPMS performance as described in this UFSAR section. 15. (Page 7.6-6). - Section 7.6.7, fifth paragraph, identified two references used to ensure ALARA issues related to the LPMS are implemented. Reference Number 7 is removed. This change is considered to be a clarification and does not affect the function or operation of the LPMS as described in the section.

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			16. (Pages 7.6-6, 7.6-7, and 7.6-8). - Section 7.6.8 provides a general functional description of the RCS Cold Overpressure Mitigation System (COMS). Several sentences were changed for clarification and/or editorial purposes. The current discussion adequately describes this mitigation system, however, the changed version is more concise. This change does not affect the mitigation logic as shown of UFSAR Figure Number 7.6-5. 17. (Pages 7.6-9, 7.6-10, and 7.6-11). - Section 7.6.9 discusses the instrumentation used for switchover from injection to recirculation after a loss-of-coolant accident (LOCA). This entire section was re-written to provide a more concise description of the instrumentation and controls. The current discussion adequately describes this feature, however, this discussion is unnecessarily wordy and several topics are duplicated in Section 7.3. This change provides a more concise description of the switchover from injection to recirculation logic and references other applicable Chapter 7 sections. There are no failure modes associated with this change. Item 8 is associated with the use of protective covers placed over specified MCR hand switches. The MCR hand switches for control valves, 1 -FCV-82-98 and -99, do not require protective covers since power is removed from the valve's power source. Thus, the protective covers are not needed for these hand switches since inadvertent actuation would not cause valve movement. Item 12 is associated with data gathering activities related to the LPMS. Specifically, a computer-based analytical system is described to use statistical and spectral analysis in order to demonstrate proper channel performance. The statistical analysis feature of this system have not provided any meaningful results. Therefore, this feature is deleted from this section. The spectral analysis feature has proven to be a valuable analysis tool. Sensors which exhibit repeatable spectral data is considered to be operational and capable of accurately converting acoustical energy to electrical signals for processing by the LPMS signal conditioning circuitry.

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			The recording and evaluation of this spectral data is considered sufficient to meet commitments discussed in this section of the UFSAR related to channel performance and channel calibration requirements. Therefore, the above described changes do not affect proper equipment/system operation and there are no credible failures associated with these changes. These changes do not impact any accidents evaluated in the UFSAR. These changes do not affect the operation of any safety related equipment/system and no credible failure modes a created or changed. Therefore, these changes do not constitute an unreviewed safety question.
2	7.7	Safety Evaluation WBPLEE-99-033 50243 1593	The Incore Flux Mapping Subsystem consists of 58 incore flux thimbles which permit measurement of the axial neutron flux distribution within the reactor core by insertion of the incore instrumentation movable miniature detectors. The Bottom Mounted Instrumentation (BMI) thimbles are inserted into the reactor through thimble guide tubes mounted on the bottom of the reactor vessel. The thimbles serve as the pressure boundary between the reactor water pressure and the atmosphere. These thimbles are subject to wear causing thinning of the thimble tube walls. This change, EDC-E-50243-A, provides a method of evaluating thimble tube wall loss and a detail for plugging tubes when it is determined that they are to be taken out of service. The BMI thimbles are in the active region of the core during normal operation and are exposed to extreme conditions of heat, RCS flow velocities and hard radiation as well as a high neutron flux. These conditions can cause wear of the thimbles which are an ANS Class II item and are a RCS pressure boundary. As such they must be monitored for wear at each cycle outage. A high degree of safety is inherent in the wall thickness of the thimbles and they will withstand RCS pressure to a wear that is equivalent to a 90% loss of wall. The amount of wear may be determined by the formula that exists. The wear formula contains an exponent that defines the shape of

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			the wear curve that is used in the predictive life of the thimbles. As the thimbles wear in accordance with the flow, flux and temperatures ranges of the plants unique core, the curve flattens out and the percentage wear for each cycle becomes smaller. This necessitates a new value for "n" that more accurately predicts the wear for the upcoming cycle. A method of determining this new "n" factor is found in Westinghouse WCAP. The calculation of a new "n" exponent does not involve a mechanism for personnel safety to be decreased nor is the plant equipment degraded in any way. Due to the ability of a thimble tube leak to be isolated and per the leak rates reported in WCAP, a leak would not be reportable as a small break LOCA. This change provides a detail to be used if a thimble tube is determined to require plugging. A compression-type cap will be installed on the thimble tubing just above the seal table, below the manual isolation valves. The cap is not subject to leakage the way a valve stem or packing may leak, thus providing more certain isolation in case of a thimble tube rupture. The tubing that has been disconnected from the seal table to allow for the cap installation will be left in place to act as a travel stop to impede thimble tube ejection in the unlikely of a mechanical failure. The tubing that is left in place may have a fitting (e.g. cap, plug, bolt) installed to provide more mass to aid in prevention of a thimble tube ejection. The passive failure of the thimble is not analyzed as a DBE. This change does not change the likelihood or the consequences of a pressure boundary failure. Therefore, this change does not create or affect any credible failures. There is no change in the function, operation, testing, maintenance or surveillance of the Incore Instrument System. The system will operate as before. The use of a compression-type cap is in accordance with design basis documents. Therefore, there is no increase in probability or consequences of evaluated accidents

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			and malfunctions, no possibility that a different type of accident or malfunction than those previously evaluated has been created, and no reduction in technical specification safety margin has occurred. Therefore, it can be seen that this change does not involve an unreviewed safety question.
2	7.7	Safety Evaluation 50475 50475 1619	DCN D-50475-A replaces the existing Anticipated Transients Without Scram (ATWS) Mitigation System Action Circuitry (AMSAC) microprocessor based logic circuitry with relay-based logic circuitry. Modifications are within the existing AMSAC cabinet and only internal components that perform the AMSAC operational logic functions are within the scope of this DCN. AMSAC inputs (turbine first stage impulse pressure input signals from AMSAC dedicated transmitters and existing narrow range steam generator level transmitters) are not changed by this DCN. Similarly, AMSAC output interfaces, including safety-related isolation devices, with the auxiliary feedwater (AFW) system and turbine trip circuitry remain unchanged. AMSAC has no primary safety functions. AMSAC performs secondary safety functions in that it is of sufficient quality and reliability to perform its intended function without contributing to transients which may challenge safety systems. AMSAC does not degrade the existing reactor protection system. The DBEs for the AMSAC are LONF/ATWS and the Loss of Load/ATWS. The scenarios of those two events are described below: A complete loss of normal feedwater occurs as a result of a malfunction in the feedwater/condensate system or its control system from such causes as the simultaneous trip of all condensate pumps, the simultaneous trip of all main feedwater pumps or the simultaneous closure of all main feedwater control, pump discharge or block valves. Because if a postulated common mode failure in the RPS, the reactor is incapable of being automatically tripped when any of several plant process variables have reached their reactor trip setpoints.

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			The most severe plant conditions that could result from a loss of load occur following a turbine trip from full power when the turbine trip is caused by a loss of main condenser vacuum. Because of a common mode failure in the protection system, the reactor is incapable of being automatically tripped as a result of the turbine trip or as the result of any several other reactor trip signals that occur later in time when several plant process variables reach their reactor trip setpoints. The components installed to perform the AMSAC logic functions have been utilized in other nuclear applications to demonstrate a history of quality and reliability. The relay-based logic circuitry has been designed that output relays shall be energized to actuate in order to prevent spurious trips and false status indication on loss of power or logic. Therefore, no new failure modes are introduced by this modification. The purpose of the AMSAC system remains the same: To function diverse from the reactor trip system to automatically start the AFW pumps and to initiate a turbine trip under conditions indicative of an ATWS. This modification does not change any system trip setpoints, and loop accuracy, as shown by calculation, is within the previous calculated loop accuracy, therefore, this modification will support the operational limits defined for AMSAC. For these reasons, this activity does not constitute a unreviewed safety question.
3	7.7	Safety Evaluation WBPLMN-01-084-0 51102 1702	This evaluation evaluates the effect of revising the control system logic to eliminate the actuation of the pressurizer backup heaters on a high level deviation signal. At present, a bistable is actuated on a high level deviation; the output of this bistable is used to both actuate an alarm on the control board and to actuate the backup heaters. The functional change is to eliminate the backup heater actuation on the bistable actuation, but to maintain the control board annunciator actuation.

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			The pressurizer level control system is used to control the RCS water inventory whenever a steam bubble is present in the pressurizer. This system also provides indications, alarms and manual controls for operator monitoring and control. The pressurizer level control system utilizes three level channels to generate signals for pressurizer level control, anticipatory alarms, indication, and recording. The level channel inputs are isolated outputs from the reactor protection system. The three level channels provide individual indication on the MCR board. During power operation, the pressurizer level setpoint is varied as a linear function of Tavg as previously described. The difference between the level signal selected for control and the reference level is compensated to eliminate any steady-state error between indicated and its setpoint. The output signal is the automatic charging flow demand signal. The uncompensated level error signal feeds two bistables. One bistable is used to provide a low level deviation alarm. This alarm is set below the programmed level by a preset amount to warn the operator that the charging system is not supplying enough charging flow to maintain pressurizer level. The second bistable actuates a high level deviation alarm if the measured level increases above the programmed level by a preset amount. This bistable also energizes the pressurizer backup heaters. The LONF/LOOP events are currently analyzed for the purpose of showing the AFW has sufficient long-term heat removal capacity. The single failure assumed for the LONF/LOOP event is failure of the turbine driven AFW pump which is the most limiting failure. Since the pressurizer backup heaters and their control circuits do not perform a primary safety function, single failure criteria does not apply. The definition of a single failure is a failure which results in the loss of a safety-related component to perform its intended safety function. Failures in systems not required to mitigate an accident are assumed when the component is in the zone of influence of the accident and is not designed and specified to remain functional in the accident environment.

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			Since there is no zone of influence for this transient, an unintended spurious operation of the backup heaters is not required to be assumed. UFSAR Section 7.7 documents that the pressurizer water level control system is not required for safety. Analysis has shown that the plant design transients are not impacted by the change, and the safety analyses are not adversely impacted. In addition, implementation of the logic change does not make any changes to the protection grade isolated inputs into the pressurizer water level control system. Therefore, based on the above, the logic modification to eliminate the automatic backup heater actuation on a high pressurizer water level deviation, and maintain manual heater actuation capability, will not adversely affect the UFSAR described design functions of any structure, system, or component (SSC). Therefore, the control system logic modification to the automatic backup heater actuation on high pressurizer water level does not require a license amendment.
N/A	N/A	Safety Evaluation WBPOSG-95-052-0 38238 N/A	DCN W-38238-B moves the ESFAS signals for 1-FCV-70-100-A and for the 6.9 KV Shutdown Boards Emergency Feeder Breakers (Diesel Generator Breakers) to other slave relays to allow testing which will not disrupt plant operation. A separate description of the change for each of these features is provided below. The DCN is a staged DCN. The stages are defined in the DCN and correspond to the completion of modifications for 1-FCV-70-100-A as one stage, and a stage for each Emergency Feeder Breaker for the 6.9 KV Shutdown Boards 1A-A, 1B-B, 2A-A, and 2B-B. This results in five stages in the DCN. a. 1-FCV-70-100-A is one of four Component Cooling System to Reactor Coolant Pump (RCP) Oil Coolers Containment Isolation Valves which are designed to close on a Containment Isolation Phase B (CI Phase B) signal. 1-FCV-70-100-A is designed such that the periodic slave relay test will cause the valve to close, isolating cooling water to the RCP Oil Coolers. Failure of the valve to open after the test would lead to a unit shutdown. The other three valves are designed such

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			that the slave relay test does not close the valves but utilizes continuity test circuits to verify slave relay operation and valve circuit integrity. This change will bring testing of 1-FCV-70-100-A in line with testing of the other three valves. This change moves the CI Phase B signal for 1-FCV-70-100-A to a different slave relay and adds a BLOCK test feature to the circuit. This will permit the slave relay test to be performed without isolating cooling water to the RCP Oil Coolers. The slave relays for these valves are tested every 92 days per Technical Specifications. This change will utilize existing abandoned cables previously used for the same function for a different containment isolation valve which was deleted. The cables have been evaluated and will be inspected for critical attributes to ensure they are acceptable for use in this application. b. The Emergency Feeder Breakers (Emergency Diesel Generator Breakers) to the 6.9 KV Shutdown Boards are designed to trip on a Safety Injection (SI) signal when the Diesel Generator (DG) is operating in the parallel test mode. The slave relays which generate the SI signal for these breakers are tested every 92 days per Technical Specifications. Performing the slave relay test for this function at power would require declaring the DGs in the train being tested inoperable and entering the applicable LCO. The Technical Specifications also require testing of the DG breaker trip on SI every 18 months with the unit shut down to demonstrate test mode override. This ensures that DG availability during accident conditions will not be degraded as a result of testing. This change moves the SI signal for the DG breakers to slave relays which are tested every 18 months, per the Technical Specifications, with the unit shut down. Justification for the 18 month frequency is provided in the TS Bases references for the existing functions of these relays. TS Change 95-090 will add a reference to DCN W-38238-B, which contains this SA/SE, in the TS Bases as the justification for this additional function. Performing the slave relay test for this function every 18

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			months will allow the test to be conducted coincident with the DG test mode override verification. This change will require relanding existing cables to different terminal blocks in the SSPS output cabinets. DCN W-38238-B provides restrictions for implementing the DG breaker circuit modification to ensure the TS requirements and FSAR commitments are met. The modification for DG breakers can be implemented in operational modes 1 through 4 provided that only one of the four power trains (1A, 2A. 1B, or 2B) is removed from service at anyone time. The power train removed from service must have the modification completed (including testing and MCR drawing update) and returned to service prior to beginning work on another power train. The modification can also be implemented in mode 5 provided that only one division of a power train (1A & 2A or 1B & 2B) is removed from service at anyone time. The division of power train removed from service must have the modification completed (including testing and MCR drawing update) and returned to service prior to beginning work on the opposite division of power train. This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. The Technical Specification is not affected. This change is in compliance with safety classification and performance requirements as specified in design basis documents. The performance of the DCN in stages does not compromise the safety of the plant. At the completion of each stage all affected components continue to meet their functional and safety requirements. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.
N/A	N/A	Safety Evaluation WBPLEE-97-065-1 39251	DCN M39251-A replaces the on-delay SI reset timing relays in the SSPS with relays which have both on-delay and off-delay. This change is being made in response to Westinghouse Infogram 96-004 which described a potential failure of the SI reset function due to "relay race." SI actuation results in energizing multiple latching slave

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		N/A	relays in the SSPS, one of which, K602, energizes the SI reset relay TD1. TD1 is a timing relay whose contacts remain open until the relay times out. TD1 contacts are interlocked with the SI reset switches in the main control room (MCR) to prevent actuating the unlatch coils of the SI slave relays and thereby prevent establishing SI block to the logic circuits. After TD1 times out (90 seconds), its contacts close, enabling the manual SI reset function from the MCR. The reactor trip breakers must also be open (P4 signal) to compete the SI block. The SI reset switch energizes the unlatch coils of the SI relays and combines with P4 to generate the SI block. The SI block removes the SI actuation signal from the logic circuits which operate the SI relays. Once the unlatch coils are energized, the K602 latch coil drops out and will not re-energize because the SI actuation signal is no longer present. The K602 contacts then open and cause TD1 to deenergize which then disables the manual SI reset circuit. If K602 drops out before the other SI relays are unlatched, they will remain latched (actuated). If this were to occur, further operation of the SI reset switch would not unlatch these relays. This problem can be averted by delaying the deenergizing of TD1 until the slave relays have had time to completely unlatch. DCN M-39251-A replaces TD1 in each of the SSPS output cabinets (Train A and Train B) with a timing relay which has both time delay on and time delay off functions. The on-delay will remain set at 90 seconds and the off delay will be set at 5 seconds. The addition of the off-delay will not prevent actuation of any safeguards loads but will permit the SI reset function to be accomplished without risk of some of the slave relays remaining in the energized state. The replacement relay will be qualified for Class 1E service. No wiring changes will be required and the SI reset operator Interface wit not be affected. The SI reset feature is provided to allow operator control of equipment after the initial phase of injection (e.g., to secure safety loads not required to mitigate the current event or to re-energize pressurizer heater control). The time delay of 90 seconds allows sufficient time for ESF loads automatically actuated by SI to reach their accident mitigation condition. The time-delay relay

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			provides capability, after the delay, to reset the SI slave relays and block the SI signal; the actuation of ESF equipment is independent of the relay. The addition of the off delay merely ensures that the reset function will operate as intended and will have no effect on operator actions that may be required after an SI. The SI signal reset function is mentioned in various sections of the FSAR (e.g., 7.3.2.2.6) and shown on FSAR Figure 7.3-3 Sheet 3. Although the 90 second time delay value for the signal reset is not given in the SAR text, the drawing on which the figure is based, 1-47W611-63-1, does show the time delay setting. The drawing and figure will be revised in accordance with plant procedures to show the addition of the off-delay to this function. The addition of the off-delay to the SI reset function will not affect initiation of safety injection or other safeguards functions nor will it impact any operator actions which may be required after an event. The change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed and no new accidents or equipment failure modes are created. The Technical 5pecifications are not affected. This change is in compliance with safety classification requirements as specified in design basis documents. The change will enhance protection system reliability since it will ensure that the SI reset function will operate as intended. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.
N/A	N/A	Safety Evaluation WBPLEE-99-108-0 50301 N/A	This change implements Phases 4 and 5 of the Integrated Computer System (ICS) upgrade plan in 34 stages. Phases 1-3 were previously implemented by DCN M-39911-A. Additional points are added to ICS including remaining Emergency Response Facilities Data System (ERFDS) data not included in previous Design Change Notice (DCN) and upgrading the inadequate core cooling monitor (ICCM) /reactor vessel level instrumentation system (RVLIS) points data link. Eagle 21

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			External Communication Interface (ECI) is added which includes a Serial to Ethernet Controller (SEC) board located in each of the Eagle 21 multi-bus chassis. The SEC is installed in the half of the chassis devoted to the Test Sequence Processor (TSP). The SEC uses the multi-bus for obtaining power only; it will not be able to communicate on the multi-bus. The SEC 'eavesdrops' on the Loop Control Processor (LCP) to TSP data link message by receiving the message in parallel with the TSP, but with no return communications. ICS software is revised to calculate points such as U1118 (Reactor Total Thermal), using digital data whenever available, however analog data will continue to be available as a default. The Main Control Room (MCR) annunciator printer is replaced with a central processing unit (CPU) and monitor. The ICS is a non-safety-related system and is isolated from safety-related equipment. Consequently, there are no safety questions associated with it. The additional points and output/display hardware changes have no safety-related functions. The Updated Final Safety Analysis Report (UFSAR) changes are minor text changes with related drawing changes. This modification introduces no increased probability of an accident or malfunction of a different type than any evaluated previously in the UFSAR. This modification introduces no increased radiological consequences of an accident or malfunction of equipment important to safety preciously evaluated in the UFSAR. Quality Related process calculations previously completed by the P2500/ERFDS are now completed by ICS. This modification will not degrade the ability of any equipment important to safety to perform its intended safety function, thereby reducing any margin of safety in the technical specifications. Therefore there is no reduction in the margin of safety as defined in the basis for any technical specification. No safety questions related to the ICS exists and therefore this activity does not constitute a license amendment.

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N/A	N/A	Safety Evaluation WBPLMN-01-013-0 50789 N/A	Various WBN design and licensing documents are currently inconsistent with regard to the maximum specified response time for containment purge isolation. In particular, two safety analysis calculations that evaluate containment purge, one for offsite dose and one for containment subcompartment differential pressure, each utilized assumptions for purge isolation response time that are less than 6 seconds now established as bounding value. To address these differences in WBN documentation regarding containment vent isolation response time, a plant specific analysis has been issued which documents that the offsite dose and containment subcompartment differential pressure analyses are bounding and conservative with an assumed 6.0 second containment purge isolation response time. The involved UFSAR accident analyses have been evaluated and revised to include the effects on containment subcompartment pressure and offsite radiological consequences of a LOCA during a containment purge. The 6.0 second total purge air valve response time will not adversely impact the results of peak clad temperature calculations during the first seconds immediately following a LOCA because the smaller mass release will not result in a containment pressure reduction at the core. The ability of the purge isolation valves to close will also not be adversely impacted during post-accident conditions. Therefore, the proposed activity will not increase the consequences of an accident previously evaluated in the UFSAR. Implementation of the design and licensing basis document changes addressed by this safety evaluation does not change the event classification of previously analyzed accidents or transients as defined in the UFSAR. The containment isolation function of purge vent isolation is a safety function and is not impacted by the changes addressed in this safety evaluation. Therefore the probability or consequences of a malfunction of equipment important to safety is not impacted. This change does not introduce new or impact existing component or system failure modes. There is no technical specification impacted, nor any new specification or surveillance created by this change. No other events addressed by the UFSAR or other licensing basis documents are impacted by this change. Therefore, the changes do not constitute a license amendment request.

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N/A	N/A	Safety Evaluation WBPLCE-99-015-0 50007 N/A	The purpose of this Safety Evaluation is to review permanent modifications to the Seismic Monitoring System. The SE also supports corresponding updates made to Section 3.7.4 of the UFSAR and to Sections 3.3.4 and B3.3.4 of the TRM to reflect the configuration of the modified Seismic Monitoring System and changes in the logic used to determine the need for shutdown following the occurrence of a seismic event that exceeds the OBE design response spectrum. DCN D-50007-A has been initiated to upgrade the Seismic Monitoring System by replacing obsolete Kinemetrics seismic monitoring instruments, removing from service all of the obsolete Engdahl instruments, and upgrading the recording capabilities of the system from analog to digital. Installation of upgraded seismic instrumentation will also permit deletion of seismic switches and triggers that are rendered redundant by the function of the upgraded system. The upgraded Seismic Monitoring System will greatly reduce the maintenance resources required to support system operation. The upgraded instrumentation will also provide seismic digital recording and analysis capabilities sufficient to permit the adoption of EPRI OBE Exceedance Criteria, which avoids unnecessary shutdown following non-damaging seismic events that exceed the OBE. The Seismic Monitoring System is *not* safety-related; nor does it have any effect on any safety related system or equipment. The upgraded seismic instrumentation will improve the accuracy and reliability of the seismic monitoring system; while at the same time greatly simplifying the maintenance and surveillance resources required to support the system. The upgrades of DCN, D-50007-A enhance the functionality of the seismic monitoring system by the addition of the Kinemetrics Condor System, which-features a high degree of redundancy and is a one-to-one, or equivalent, replacement for the existing SMA-3/SMP-1 based recording and playback system. The inclusion of an onboard processing computer and strong motion analysis software as part of the Condor System facilitates timely evaluation of the recorded

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			event. The removal from service of the obsolete Engdahl instruments greatly reduces the maintenance and surveillance requirements for the Seismic Monitoring System without any significant loss of data capability. The use of EPRI OBE Exceedance Criteria ensures that the decision for a controlled shutdown is based on actual damage potential of the event, which reduces the risk associated with unnecessary shutdowns. UFSAR Section 3.7.4 and TRM Sections 3.3.4 and B3.3.4 have been revised to reflect the upgraded configuration of the Seismic Monitoring System and the adoption of EPRI OBE Exceedance Criteria for shutdown logic. The upgraded Seismic Monitoring System and use of EPRI OBE Exceedance Criteria maintain the UFSAR commitment to the intent of Reg Guide. 1.12, Revision 1. The Seismic Monitoring System is *not* safety-related; nor does it have any effect on safety-related systems or equipment. The upgraded seismic instrumentation will improve the accuracy and reliability of the seismic monitoring system; while at the same time greatly simplifying the maintenance and surveillance resources required to support the system. The upgrades of DCN D-50007-A enhance the functionality of the seismic monitoring system by the addition of the new instrumentation, which features a high degree of redundancy and is a one-to-one, or equivalent, replacement for the existing analog based recording and playback system. The inclusion of an onboard processing computer and strong motion analysis software facilitates timely evaluation of the recorded event. The removal from service of the obsolete instruments greatly reduces the maintenance and surveillance requirements for the Seismic Monitoring System without any significant loss of data capability. The use of EPRI OBE Exceedance Criteria ensures that the decision for a controlled shutdown is based on actual damage potential of-the event, which reduces the shutdown risk associated with unnecessary shutdowns. Therefore, implementation of DCN D-50007-A and corresponding revisions to the UFSAR and TRM do not involve a

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			unreviewed safety question.
N/A	N/A	Safety Evaluation WBPLEE-98-055-0 39973 N/A	Design Change Notice (DCN) S-39973-A resolves several unrelated Drawing Deviations so that the drawings more accurately reflect the as-constructed plant configuration and to ensure that design documents are consistent. No hardware or functional changes are being made by this DCN. This DCN corrects drawing discrepancies in accordance with Drawing Deviation numbers 98-0041, 98-0040, 99-0039, and 99-0036. Specifically, this DCN makes the following changes: Revise electrical control diagrams 1-47W610-90-4 And 1-47W610-30-4 (FSAR Figure 9.4-17) to remove the flow signal from 0-EM-90-300/C to ERFDS and the P2500 computer as shown on 1-47W610-30-4. The flow signal output to ERFDS and the P2500 computer is already shown on 1-47W610-90-4, and 0-EM-90-300/C does not provide two outputs to these computer based system. Additionally, electrical control diagram 1-47W610-90-4 is revised to change P2500 computer input point from 2704A to F2704A. Revise electrical control diagram I-47W610-30-2 (FSAR Figure 9.4-31) to indicate that TE-30-210Q through TE-30-210AH input to the P2500 computer instead of a recorder. Revise electrical connection diagram 45N1678-1 to show the “to” designation for cable 1C1182 as 1-CMPT-261-R158 instead of 1-CMPT-264-R158. Revise the Cable and Conduit Routing System (CCRS) for cable 1C1181 to indicate the system as 261, the “from ID” as 1-CMPT-261-R158, and the “to” drawing as 45N1678-5.

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			Revise electrical schematic 1-45W600-46-1 to correct the fuse identification for auxiliary relay rack 1-R-72, and to provide a table to identify fuses used in the stop valve circuits for both MFPT 1A and MFPT 1B. Revise electrical control diagram 1-47W610-47-2 (FSAR Figure 10.2-3) to show the correct electrical overspeed trip setpoint of 111% of rated speed as described in FSAR Section 10.2.2. Additionally, revise the "Gen Bkr Open" trip for Turbine Trip Bus "A" and "B" to indicate that there is no dependence on time. Revise electrical schematic 1-45W600-47-2 (FSAR Figure 10-2- 1) and System Description Document, N3-47-4002, to correctly identify the function of 1-LS-47-105 as sensing lube oil tank level. Since these are documentation changes only and do not represent any functional, operational, or physical change to the plant, the minor changes to the above FSAR figures by this DCN do not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the FSAR. In addition, the documentation changes do not create the possibility of an accident or equipment malfunction of a different type than previously evaluated and does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, the changes made by DCN S-39973-A do not constitute an unreviewed safety question.
N/A	N/A	Safety Evaluation WBPLEE-98-069-0 50003 N/A	The pressurizer relief tank (PRT) condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from specific relief valves located inside and outside the containment is also piped to the PRT. The expected leakage from various reactor coolant pressure boundary components also goes to the PRT. The high temperature alarm is intended to warn the operator of RCS leakage into the

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			PRT. During hot weather and minor temperature excursions in containment, the ambient temperature in the vicinity of the PRT approaches and sometimes exceeds the setpoint of the high temperature alarm. Thus, the alarm comes into the control room even though the liquid temperature of the PRT has not increased due to RCS leakage. This causes a nuisance alarm in the main control room and masks any increase in temperature in the PRT that is due to the relieving of the RCS to the PRT. Increasing the temperature setpoint would prevent this from happening. This design change, DCN-D-50003-A, revises the high temperature setpoint and normal operating range for level in the PRT. The current high temperature setpoint is 112.5 degrees F and will be recalibrated to 120 degrees F. The current operating range for level is 55.5 (low level) to 80 (high level) IN H20. This range will also be recalibrated to 87 to 80 IN H2O. When the temperature or level in the PRT exceeds the setpoint, an alarm in the main control room is actuated and the operator takes the appropriate corrective action. The existing alarm setpoints for PRT temperature and level are also being revised in ERFDS and the P2500 Computer. If there is an accidental depressurization of the reactor coolant system (UFSAR 15.2.12) due to an inadvertent opening of a pressurizer safety or relief valve, the discharge from the valve will go to the pressurizer relief tank. However, the PRT is not a component that is important to safety and is not required to mitigate this design basis accident and, therefore, the failure of the PRT is inconsequential to nuclear safety. This change is revising the setpoints for the level alarms because the increase in temperature of the water reduces the cooling capacity of the PRT from that at the lower temperature alarm setpoint (112.5 degrees F). Westinghouse letter, WAT-D-10558, establishes the new level alarm setpoints; for the PRT. With the new level alarm setpoints, the PRT will have the same cooling capacity as it did prior to the

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			change. There is no change in the function, operation, testing, maintenance or surveillance of the affected components. The system will operate as before. Therefore, there is no increase in probability or consequences of evaluated accidents and malfunctions, no possibility of a different type of accident or malfunction than those previously evaluated has been created, and no reduction in Technical Specification safety margin has occurred. Therefore, it can be seen that this change does not involve an Unreviewed Safety Question.
N/A	N/A	Safety Evaluation WBPLEE-00-010-0 50484 N/A	DCN M-39816-B, previously issued and closed, provided for the design and installation of the Supplemental Condenser Circulating Water (SCCW) sub-system addition to the Condenser Circulating Water system. The SCCW system supplies water from the Watts Bar Reservoir to provide a source of cooler water to the existing Unit 1 cooling tower discharge flume. SCCW discharge line flow and temperature are monitored and recorded via a solar powered data logger which records the output from a flow/temperature monitoring annubar assembly mounted in the discharge line. This data is recorded to provide proof of environmental compliance of the discharge flow. DCN D-50437-A, previously issued and closed, addressed the design and implementation of the changes to the ICS that were necessary to receive the SCCW thermal discharge monitoring system data from the Environmental Data Station (EDS) computer over the existing EDS to ICS computer data link, store the date, and display the data. One of the monitoring stations is the SCCW effluent monitoring station (Station 31) which provides temperature and flow data from the SCCW discharge line annubar sensor. This change, DCN D-50484-A, abandons in place the SCCW discharge line flow transmitter and temperature element, and removes the solar panel and battery box. This DCN also modifies the ICS NPDES view to remove the reference to the SCCW Effluent Station, temperature element TE-27-111, and flow element FE-27-111. The ICS will continue to receive the SCCW effluent temperature and effluent flow data over the EDS to ICS data link: the difference being

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			that the data comes from instrumentation at the Glory Hole (Station 32) which is outside configuration control. The SCCW sub-system does not perform any safety related function. There are no design bases accidents affected by this DCN. The SCCW discharge flow and temperature data will continue to be recorded via instrumentation outside design control. It is unlikely that the flow/temperature monitoring and recording system would be down for an extended period and it is unlikely that a potential violation of the permit would occur. The risk of this unlikely occurrence is considered acceptable. Therefore, there will be no credible failure modes introduced by this DCN. The equipment being removed from configuration control performs no safety function, and does not interface with any equipment important to safety. The proposed design change does not increase the probability of an accident or the occurrence of a malfunction of equipment important to safety. The consequences of an accident or a malfunction of equipment will not be increased. No accidents or malfunctions of a different type than previously evaluated in the UFSAR are created. The proposed design change does not affect any technical specification or margin of safety identified in the technical specification bases. Therefore, this change does not involve any unreviewed safety question.
N/A	N/A	Safety Evaluation WBPLEE-99-088 50388 N/A	1-XIS-68-70 is a hydraulic isolator for 1-PT-68-70, which measures Reactor Coolant System pressure at the hot leg for loop four and is one of the Post Accident Monitoring channels. The isolator and transmitter utilize a sensor bellows and capillary tubing to sense the system pressure. The sensor bellows shares a sense line with pressure transmitter, 1-PT-68-68, and pressure switches 1-PS-68-68A, -68G/H, and -68H/G which are inputs into the Cold Overpressure Mitigation System (COMS). However, there is no means to isolate the sensor bellows without isolating the COMS instrumentation. During the Reactor Coolant System vacuum refill the pressure transmitter 1-PT-68-68 and the pressure switches are required in order for the Cold Overpressure Mitigation System to be operable. However, pulling a vacuum

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			on the bellows for 1-XIS-68-70/1-PT-68-70 will cause damage. This change, DCN-D-50388-A, will install an isolation valve so the COMS instrumentation can be in service while the isolator bellows is protected during the Reactor Coolant System vacuum refill. During normal operation the valve will be open and 1-XIS-68-70/1-PT-68-70 will perform its function. During a refueling outage, when 1-XIS-68-70/1-PT-68-70 is not required, the valve will be closed to protect the bellows during vacuum refill of the Reactor Coolant System. There is no change in the function, operation, testing, maintenance or surveillance of the affected components. The system will operate as before. Therefore, there is no increase in probability or consequences of evaluated accidents and malfunctions, no possibility of a different type of accident or malfunction than those previously evaluated has been created, and no reduction in Technical Specification safety margin has occurred. Therefore, it can be seen that this change does not involve an Unreviewed Safety Question.

ENCLOSURE 3

TVA LETTER DATED SEPTEMBER 9, 2010
WATTS BAR NUCLEAR PLANT (WBN) UNIT 2
REQUEST FOR ADDITIONAL INFORMATION REGARDING FSAR
RELATED TO INSTRUMENTATION AND CONTROLS (TAC NO. ME2371)

LIST OF UNIT 1 UFSAR CHAPTER 7.0 CHANGES
WHICH REQUIRED A LICENSE AMENDMENT

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1	7.2	WBPLEE-98-036-1 M 39265-A 1512	Deleted the power range high neutron flux negative rate trip. NRC approved this change in Amendment 18 of the Technical Specifications.
4	7.1	WBNLEE-03-041 DCN 51297-A 1769	Reactor Coolant System Flow Rate Measurement Using Elbow Tap Methodology - NRC approved this change in Amendment 47 of the Technical Specifications.
6	7.7	N/A N/A 1882	Alternative Means for Monitoring Control or Shutdown Rod Position - NRC approved this change in Amendment 58 of the Technical Specifications.