Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-20-030

April 29, 2020

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
Facility Operating License No. NPF-96
NRC Docket No. 50-391

Subject: **Response to Request for Additional Information to License Amendment Request for Measurement Uncertainty Recapture Power Uprate (WBN-TS-19-06) (EPID L-2019-LLS-0000)**

References:

1. TVA Letter to NRC, CNL-19-082, "License Amendment Request for Measurement Uncertainty Recapture Power Uprate (WBN-TS-19-06)," dated October 10, 2019 (ML19283G117)

2. NRC Electronic Mail to TVA, "Request for Additional Information for WBN2 Request Measurement Uncertainty Recapture Power Uprate (L-2019-LLS-0000)," dated March 24, 2020 (ML20084M194)

3. NRC Electronic Mail to TVA, "Request for Additional Information for WBN2 Request Measurement Uncertainty Recapture Power Uprate (L-2019-LLS-0000) - Part 2," dated March 26, 2020 (ML20086G480)

In Reference 1, Tennessee Valley Authority (TVA) submitted a request for an amendment to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant (WBN), Unit 2. The proposed license amendment request (LAR) would increase the WBN Unit 2 authorized core power level from 3411 megawatts thermal (MWt) to 3459 MWt (i.e., an increase of approximately 1.4% Rated Thermal Power), based on the use of the Caldon®[1] Leading Edge Flow Meter (LEFM®[1]) CheckPlus System.

In References 2 and 3, the Nuclear Regulatory Commission (NRC) provided a request for additional Information (RAI) and requested that TVA respond by April 30, 2020. Enclosure 1 to this letter provides the response to the RAI.

---

[1] Caldon, Inc. is now part of the Measurement Systems Division of Cameron International Corporation (Cameron). Caldon and LEFM are registered trademarks of Cameron.

Additionally, this letter addresses an error in Reference 1. Specifically, items 3 and 4 to Section 2.1, "Description of the Proposed Change," of Reference 1, and the proposed markups to WBN Unit 2 TS 5.9.5b in Enclosures 3 and 4 to Reference 1 referred to document number 10 instead of document number 11, which is consistent with the proposed markup to WBN Unit 2 Technical Specification (TS) 5.9.5b.11, "CORE OPERATING LIMITS REPORT (COLR)," in Enclosures 3 and 4 to Reference 1.

Enclosure 2 to this letter provides revised items 3 and 4 to Section 2.1 to Reference 1. Enclosure 3 to this letter provides the revised mark-up to WBN Unit 2 TS 5.9.5b and Enclosure 4 to this letter provides the revised re-typed WBN Unit 2 TS 5.9.5b to correct the reference from document number 10 to document number 11. Enclosures 3 and 4 to this letter supersede the proposed changes to WBN Unit 2 TS 5.9.5b in Enclosures 3 and 4 to Reference 1. In response to NRC RAI SNSB-RCS-1, Enclosure 5 provides a proposed update to Table 15.1-2 of the WBN dual-unit Updated Final Safety Analysis Report for NRC information only.

This letter does not change the conclusions, the no significant hazards consideration, nor the environmental considerations contained in Reference 1. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the enclosures to the Tennessee Department of Environment and Conservation.

There are no new regulatory commitments associated with this submittal. Please address any questions regarding this request to Kimberly D. Hulvey, Senior Fleet Licensing Manager, at (423) 751-3275.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 29th day of April 2020.

Respectfully,

James Barstow

James Barstow
Vice President, Nuclear Regulatory Affairs & Support Services

Enclosures:

1. Response to NRC Request for Additional Information
2. Revised Description of the Proposed Change
3. Revised Proposed WBN Unit 2 TS 5.9.5.b (Markup)
4. Revised Proposed WBN Unit 2 TS 5.9.5.b (Re-typed)
5. Proposed Update to Table 15.1-2 of the WBN dual-unit UFSAR (For Information Only)

cc (Enclosures):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Watts Bar Nuclear Plant
NRC Project Manager – Watts Bar Nuclear Plant
Director, Division of Radiological Health – Tennessee State Department of Environment and Conservation

NRC Introduction

By letter dated October 10, 2019 (Agencywide Documents Access and Management System *(ADAMS) Accession No. ML19283G117) to the U.S. Nuclear Regulatory Commission (NRC), Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) for the Watts Bar Nuclear Plant, Unit 2 (WBN2). The proposed amendment would increase the authorized core power level by approximately 1.4 percent rated thermal power from 3411 megawatts thermal (MWt) to 3459 MWt. Additionally, the proposed amendment would revise Technical Specification (TS) 1.1, "Definitions," and TS 5.9.5b, "Core Operating Limits Report (COLR)," to reflect changes to the power level and use of the leading edge flowmeter (LEFM).*

NRC RAI SNSB-Containment-1:

To meet General Design Criterion (GDC) 50, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 6.2.1.3 specifies that the reactor power should be considered when the mass and energy release from the break is under evaluation (ADAMS Accession No. ML053560191). Please provide the power level (reactor core or NSSS) for each of the following analyses for both current licensing and the proposed 1.4% measurement uncertainty recapture (MUR) power uprate conditions:

(a) Short-Term LOCA Mass and Energy Release Analysis
(b) Loop Subcompartment Analysis
(c) Reactor Cavity Analysis
(d) Pressurizer Enclosure Analysis
(e) Maximum Reverse Pressure Differential Analysis

TVA Response

The short-term loss of coolant accident (LOCA) mass and energy release analysis is not a stand-alone analysis. The mass and energy releases are unique to each pipe break and location for each short-term subcompartment analysis. Therefore, no power level can be provided for Item (a). The power level assumed for the generation of the LOCA mass and energy releases used for the compartment response for the current design basis analyses are provided in Table RAI SNSB-Cont-1. The core power including uncertainty that pertains to the evaluations performed for the 1.4% MUR is 1.006% x 3459 MWt = 3479 MWt. Based on the comparison of the current analyses of record analyzed power level to the proposed MUR power level, the core power including uncertainty is analytically equivalent for the loop compartment analysis, the reactor cavity analysis, and the pressurizer enclosure analysis. The power level used for the maximum reverse pressure differential analysis current analysis of record bounds the proposed 1.4% MUR power level.

Table RAI SNSB-Cont-1 – Analyzed Power Level (includes uncertainty)

Analysis	Current Analysis of Record
Loop Compartment	102% of 3411 MWt = 3479 MWt
Reactor Cavity	102% of 3411 MWt = 3479 MWt
Pressurizer Enclosure	102% of 3411 MWt = 3479 MWt
Maximum Reverse Pressure Differential	102% of 3570 MWt = 3641 MWt

NRC RAI SNSB-Containment-2:

TVA stated that the loop subcompartment analysis is based on the use of Zaloudek correlation to calculate the subcooled water release. The current licensing basis reactor coolant temperatures for the Zaloudek correlation calculation are 555.2 °F and 617.1 °F for core/vessel inlet and outlet, respectively. The reactor coolant temperatures for 1.4% MUR power uprate are 557.3 °F and 619.1 °F core/vessel inlet and outlet, respectively. TVA stated that the use of lower reactor coolant temperatures will lead to higher critical mass flux from the reactor coolant system break. Hence, TVA determined that the current licensing basis mass and energy release would bound the mass and energy release for the 1.4% MUR power uprate.

However, TVA also stated in the Reactor Cavity Analysis section in the LAR that the 1.4% MUR power uprate would increase the break's critical mass flux by 3.6% as based on the Zaloudek correlation. Apparently, for the same application of Zaloudek correlation with the same reactor coolant temperature at reactor vessel inlet break, there exists contradictive conclusions in the comparison of critical mass flux between these two reactor power conditions (i.e., current licensed power versus 1.4% MUR power uprate) from these two analyses (i.e., loop subcompartment versus reactor cavity).

SRP Section 6.2.1.4 applies the GDC 50 requirements to postulated line break to assure that the mass and energy release should be appropriately determined first (ADAMS Accession No. ML070620010). Please explain and resolve the contradictive determinations for the mass and energy release.

<u>TVA Response</u>

The purpose of the short-term subcompartment analyses is to show that the interior containment walls and structures can withstand the maximum calculated differential pressure created by the pressure pulse resulting from a postulated rupture of primary or secondary piping. In order to achieve that maximum calculated differential pressure, the mass released from the postulated break needs to be maximized. The parameters that have been determined to show the greatest sensitivity to maximizing the mass releases and the direction of conservatism for each parameter are as follows:

- Initial reactor coolant system (RCS) pressure – maximum possible
- Initial RCS temperature – minimum possible
- Break area – maximum possible

The Zaloudek correlation is part of the approved methodology for generating the short-term LOCA mass and energy releases [Section III of WCAP-8312-A, Revision 2 (Reference)]. For evaluation purposes, the Zaloudek correlation can be used to conservatively evaluate the impact of changes in the RCS temperature at the break location. The RCS operating temperatures that pertain to the current analysis of record reactor cavity analysis are not the same as the loop compartment analysis of record so there is not any contradiction in the assessments for the loop compartment analysis and the reactor cavity analysis.

The values used for the reactor cavity breaks are an inlet temperature of 559 degrees Fahrenheit (°F) and an outlet temperature of 624°F. These values are higher than the proposed MUR values of 557.3°F and 619.1°F. To clarify, the temperatures used for the loop compartment analysis of record and the reactor cavity analysis of record are presented in Table RAI SNSB-Cont-2 along with the proposed values for the WBN Unit 2 1.4% MUR program.

Table RAI SNSB-Cont-2		
Analysis	**Current Analysis of Record Conditions**	**Conditions Proposed for 1.4% MUR**
Loop Compartment Breaks		
Thot	617.1°F	619.1°F
Tcold	552.2°F	557.3°F
Reactor Cavity Breaks and Pressurizer Spray Line Break		
Thot	624°F	619.1°F
Tcold	559°F	557.3°F

With the goal of maximizing the mass flux, Table RAI SNSB-Cont-2 demonstrates that the current analysis of record temperatures for the reactor cavity breaks would not achieve this. Therefore, the reactor cavity releases were initially assessed a penalty of 3.6 percent (%) solely due to the effect of RCS temperature differences on the mass flux determined from the Zaloudek correlation.

Subsequently, the postulated break size in the reactor cavity region was investigated for possible margin. The current design basis analysis used a break area of 127 square inches (in^2) while a plant specific pipe motion/displacement analysis estimated a break size for the as-built plant at the reactor vessel inlet and outlet nozzles is less than 45 in^2. The factor of 2.8 reduction in the break size more than offsets the 3.6% increase in the reactor cavity analysis mass and energy releases due to a temperature reduction. Therefore, the current design basis reactor cavity analysis LOCA mass and energy releases remain conservative for the 1.4% MUR program.

Reference

"Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, Revision 1, WCAP-8264-P-A (Proprietary), WCAP-8312 (Nonproprietary), dated August 1975

NRC RAI SNSB-Containment-3:

The worst break possible in the pressurizer enclosure, as described in Final Safety Analysis Report, as updated (UFSAR) Section 6.2.1.3.9 (ADAMS Accession No. ML19336A067), is a double-ended rupture of the 6-inch (approximate area of 0.1963 square feet (ft^2)) spray line. The rupture is assumed to occur at the top of the enclosure. However, TVA stated that the as-built break is located at either cold leg spray nozzle or pressurizer spray nozzle with areas of 0.0645 ft2 or 0.08727 ft2, respectively.

SRP Section 6.2.1.2 applies the GDC 4 requirements to postulated line break to assure that the compartment structure and systems would be protected from the impact of a high energy line break (ADAMS Accession No. ML070620009). Please provide justification and the supporting analysis for the change of break from the 6-inch spray line to pressurizer spray nozzle. Include an explanation for why the break is assumed to only occur at the nozzle, or that a break cannot occur in the pressurizer spray line.

TVA Response

The original design basis analysis for the transient for the double-ended pressurizer spray line break within the pressurizer enclosure used LOCA mass and energy releases for a six-inch inside diameter hole in the cold leg. The hole was assumed to have an area of 0.1963 square feet (ft^2) in the current analysis of record. The area that was assumed for a four-inch inside diameter hole in the top of the pressurizer to represent the pressurizer spray nozzle was 0.08727 ft^2. No spray line piping was modeled so there was not any resistance associated with the length of the piping from the cold leg to the top of the pressurizer, or any control valves, tee fittings, or other flow losses that could reduce the break flow from either side of the break. The as-built pressurizer spray line piping at WBN Unit 2 is a four-inch schedule 160 pipe segment connected to the cold leg which has an area of 0.0645 ft^2 and the pressurizer spray nozzle attached to the upper head of the pressurizer is a four-inch, schedule 160 nozzle. The inside diameter of the spray nozzle in the upper head of the pressurizer would control the break from the pressurizer side of a double-ended break regardless of where the spray line pipe is postulated to break. The inside diameter of the pressurizer spray line piping would control the break from the cold leg side of the break regardless of where the pipe is postulated to break. The break area on the cold leg side of the break would be reduced by greater than a factor of three (i.e., 0.1963/0.0645 = 3.04) and the break area on the spray nozzle side of the break would be reduced by greater than a factor of 1.3 (i.e., 0.08727/0.0645 = 1.35). The reduction between the analyzed break sizes and the as-built piping sizes offsets the 3.6% penalty determined due to the differences in the RCS temperatures from the current design basis analysis. Therefore, the short term LOCA mass and energy releases for a nominal six-inch diameter pipe on one side of the break and a nominal four-inch diameter nozzle on the other side of the break remain conservative for a double-ended pressurizer spray line break postulated to occur anywhere in the pressurizer spray line piping for WBN Unit 2.

NRC RAI SNSB-RCS-1:

Page E2-15 of Enclosure 2 to the LAR indicated that for events that are departure from nucleate boiling (DNB) limited or for that Revised Thermal Design Procedure (RTDP) is used, the transient analyses assumed 3475 MWt as initial power level (representing the nominal uprated power of 3459 MWt (101.4% of 3411 MWt), plus a reactor coolant pump (RCP) net heat input of 16 MWt). For events that are not DNB limited or for that the RTDP was not applied, the analyses were performed at initial conditions obtained by adding the bounding steady-state errors to nominal values in such a manner to maximize the impact on the limiting parameter. TVA described each analysis briefly for the UFSAR Chapter 15 events in Item II.1.D.iii of Enclosure 2 to the LAR and provided in Table II.1-1 the power levels assumed in the analyses of the UFSAR Chapter 15 events to support the MUR power uprate application. TVA indicated that the analyses of record (AORs) for UFSAR Chapter 15 events were performed at power levels equal to or greater than the MUR uprated power level and claimed that the AORs reflected in the WBN2 UFSAR were unaffected by the MUR power level and remained acceptable for WBN2 MUR power uprate to meet the requirements of GDCs 10, 15 and 10 CFR 50.46.

The NRC staff compared the power levels shown in Table II.1-1 with that in the most recent version of WBN2 UFSAR TABLE 15.1-2, "Summary of Initial Conditions and Computer Codes Used," (ADAMS Accession No. ML19176A135) and found that that the power levels in Table II.1-1 assumed in the analyses supporting the MUR power uprate application are equal to or greater than that listed in WBN UFSAR Table 15.1-2 for the analyses of most events. In Table 1 below, the NRC staff identified the power levels that were different for the events listed in WBN UFSAR Table 15.1-2 for the AOR and Table II.1-1 in the LAR. To aid the review, NRC staff is requesting the following items:

a. *Confirm that the power levels for the analysis of the events listed in WBN UFSAR Table 15.1-2 have been updated to represent the AOR of WBN (at the MUR power level).*

b. *If the events in Table 15.1-2 were not updated, provide the updated NRC-approved UFSAR Table 15.1-2 to demonstrate that the power levels assumed in the AORs bound the power level for the MUR power uprate application.*

c. *If the events in Table 15.1-2 were not analyzed at the MUR power level, provide the results of the reanalyses for those events. Otherwise, for each of those reanalyses previously approved by the NRC, provide a reference of the NRC safety evaluations approving the reanalyses.*

Table 1 Power Levels Used in the UFSAR Chapter 15 Analyses

Event No. (Shown in Table II.1-1 of Enclosure 2) UFSAR Section No. Event Title	***Analytical Power Level (MWt) from Table II.1-1 of Enclosure 2 (ADAMS 19283G119)***	***Analytical Power Level MWt) from WBN UFSAR TABLE 15.1-2 (ADAMS ML19176A135)***
(2) UFSAR 15.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	*3475 (1.014 % of 3411 MW plus 16 MW RCP heat)*	*3425*
(3) UFSAR 15.2.3 Rod Cluster Control Assembly Misalignment	*3475*	*3425*
(4) UFSAR 15.2.4 Uncontrolled Boron Dilution	*0 and 3475*	*0 and 3425*
(8) UFSAR 15.2.8 Loss of Normal Feedwater	*3479 (1.02 % of 3411 MW plus 16 MW RCP heat*	*3475*
(9) UFSAR 15.2.9 Coincident Loss of Onsite and External (Offsite) AC Power to the Station – Loss of Offsite Power to the Station Auxiliaries	*3475*	*This event is not included in the FSAR Table*
(10) UFSAR 15.2.10 Excessive Heat Removal Due to Feedwater System Malfunctions	*3475*	*3425*

Event No. (Shown in Table II.1-1 of Enclosure 2) ***UFSAR Section No.*** ***Event Title***	***Analytical Power Level (MWt) from Table II.1-1 of Enclosure 2*** ***(ADAMS 19283G119)***	***Analytical Power Level MWt) from WBN UFSAR TABLE 15.1-2*** ***(ADAMS ML19176A135)***
(11) *UFSAR 15.2.11* *Excessive Load Increase Incident*	*3475*	*Not Available (NA)*
(12) *UFSAR 15.2.12* *Accidental Depressurization of the Reactor Coolant System*	*3475*	*3425*
(15) *UFSAR 15.2.15* *Chemical and Volume Control System Malfunction During Power Operation*	*3475*	*NA*
(16) *UFSAR 15.3.1* *Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System*	*3480*	*3475*
(18) *UFSAR 15.3.3* *Inadvertent Loading of a Fuel Assembly into an Improper Position*	*3425*	*3425*
(21) *UFSAR 15.3.6* *Single Rod Cluster Control Assembly Withdrawal at Full Power*	*3475*	*3425*
(22) *UFSAR 15.4.1* *Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)*	*3479.8*	*3475*
(24) *Not in UFSAR* *Steam Line Break with Coincident Rod Withdrawal at Power*	*3475*	*NA*
(25) *UFSAR 15.4.2.2* *Major Rupture of a Main Feedwater Pipe*	*3475*	*3425*

Event No. (Shown in Table II.1-1 of Enclosure 2) UFSAR Section No. Event Title	***Analytical Power Level (MWt) from Table II.1-1 of Enclosure 2 (ADAMS 19283G119)***	***Analytical Power Level MWt) from WBN UFSAR TABLE 15.1-2 (ADAMS ML19176A135)***
(26) UFSAR 15.4.3 Steam Generator Tube Rupture	*3475*	*3427*
(29) UFSAR 15.4.6 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	*0 and 3475*	*0 and 3411*
(30) Not in UFSAR Anticipated Transients Without Scram	*3479 (102% of 3411)*	*NA*

TVA Response

a) The power levels for the analysis of the events listed in WBN UFSAR Table 15.1-2 have not been updated to represent the AOR of WBN (at the MUR power level) because the WBN Unit 2 MUR power uprate LAR (Reference 1) has not yet been approved by the NRC. However, footnote 6 in UFSAR Table 15.1-2 notes that the analyses for Unit 1 support the MUR power level either directly or through evaluation.

b) Enclosure 5 to this submittal contains a proposed revision to Table 15.1-2 of the WBN UFSAR reflecting the MUR power level for both units, which demonstrates that the power levels assumed in the AORs bound the power level for the MUR power uprate application. The proposed revision to the Table 15.1-2 of the WBN dual-unit UFSAR will be implemented following NRC approval of Reference 1.

c) The MUR power uprate was approved for WBN Unit 1 in Reference 2, which increased the full core thermal power rating from 3411 MWt to 3459 MWt. Reference 2 was subsequently incorporated into Amendment 2 of the WBN Unit 1 UFSAR (Reference 3), which reflected operation at the MUR core power level of 3459 MWt and nuclear steam supply system (NSSS) power level of 3475 MWt.

   In Reference 4 regarding the reactivation of construction activities for WBN Unit 2, TVA stated, "alignment of the WBN Unit 1 and 2 licensing and design bases will ensure that there is operational fidelity between the units and at the same time demonstrate, and ensure that WBN Unit 2 complies with applicable NRC regulatory requirements." As noted in Reference 5, which outlined the regulatory framework for the completion of Watts Bar Unit 2, "The current licensing basis for Unit 1 will be used as the reference basis for the review and licensing of Unit 2."

   Accordingly, the MUR analyses approved by the NRC in Reference 2 were reflected in Amendment 97 of the WBN Unit 2 FSAR (Reference 6), which updated UFSAR Table 15.1-2 to reflect the MUR power level for certain reanalyzed events (e.g., partial loss of flow, loss of normal feedwater/loss of offsite power, complete loss of flow, large break LOCA, locked rotor) and added footnote 5 which stated, "Several of these analyses are conservatively based upon a core power of 3459 MWt and NSSS power of 3475 MWt, based upon a redefinition of the 2% power uncertainty (2% to 0.6%), which bounds a core power of 3411 MWt and NSSS power of 3425 MWt."

Therefore, the WBN Unit 2 analyses in Table 15.1-2 of the WBN UFSAR support the MUR power level consistent with the WBN Unit 1 analyses either directly or through evaluation. Table SNSB-RCS-1 lists where each of the analyses identified in Table 1 of the RAI were specifically reviewed by the NRC.

Table SNSB-RCS-1

Event No. (Shown in Table II.1-1 of Enclosure 2) **UFSAR Section No.** **Event Title**	**Reference to Where NRC Reviewed the Analyses**
(2) UFSAR 15.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Page 7 of the SE of Reference 2 Section 15.2.4.2 of Reference 7 Section 15.2.4.2 of Reference 10
(3) UFSAR 15.2.3 Rod Cluster Control Assembly Misalignment	Page 8 of the SE of Reference 2 Section 15.2.4.3 of Reference 7 Section 15.2.4.3 of Reference 10
(4) UFSAR 15.2.4 Uncontrolled Boron Dilution	Page 9 of the SE of Reference 2 Section 15.2.4.4 of Reference 7 Section 15.2.4.4 of Reference 11
(8) UFSAR 15.2.8 Loss of Normal Feedwater	Pages 8 and 9 of the SE of Reference 2 Section 15.2.1 of Reference 7 Section 15.2.1.3 of Reference 10
(9) UFSAR 15.2.9 Coincident Loss of Onsite and External (Offsite) AC Power to the Station – Loss of Offsite Power to the Station Auxiliaries	Section 15.2.1 of Reference 7 Section 15.2.1.4 of Reference 10
(10) UFSAR 15.2.10 Excessive Heat Removal Due to Feedwater System Malfunctions	Page 8 of the SE of Reference 2 Section 15.2.2 of Reference 7 Section 15.2.2.2 of Reference 10
(11) UFSAR 15.2.11 Excessive Load Increase Incident	Page 8 of the SE of Reference 2 Section 15.2.2 of Reference 7 Section 15.2.2.3 of Reference 10
(12) UFSAR 15.2.12 Accidental Depressurization of the Reactor Coolant System	Page 8 of the SE of Reference 2 Section 15.2.3 of Reference 7
(15) UFSAR 15.2.15 Chemical and Volume Control System Malfunction During Power Operation	Page 7 of the SE of Reference 2 Section 15.2.3.1 of Reference 10

Table SNSB-RCS-1	
Event No. (Shown in Table II.1-1 of Enclosure 2) **UFSAR Section No.** **Event Title**	**Reference to Where NRC Reviewed the Analyses**
(16) UFSAR 15.3.1 Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System	Page 7 of the SE of Reference 2 Sections 15.3.1 and 15.4.1 of Reference 7 Section 15.3.1 of Reference 10 (also see footnote 5 on page 15-21)
(18) UFSAR 15.3.3 Inadvertent Loading of a Fuel Assembly into an Improper Position	Section 15.2.4.5 of Reference 7 Section 15.2.4.5 of Reference 10
(21) UFSAR 15.3.6 Single Rod Cluster Control Assembly Withdrawal at Full Power	Page 8 of the SE of Reference 2 Section 15.2.4.6 of Reference 7 Section 15.2.4.6 of Reference 10
(22) UFSAR 15.4.1 Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)	Page 6 of the SE of Reference 2 Sections 15.3.1 and 15.4.1 of Reference 7 Section 15.3.1 of Reference 10
(24) Not in UFSAR Steam Line Break with Coincident Rod Withdrawal at Power	Page 8 of the SE of Reference 2 Section 15.3.2 of Reference 10
(25) UFSAR 15.4.2.2 Major Rupture of a Main Feedwater Pipe	Pages 8 and 9 of the SE of Reference 2 Section 15.3.3 of Reference 7 Section 15.3.3 of Reference 10
(26) UFSAR 15.4.3 Steam Generator Tube Rupture	Page 6 of the SE of Reference 2 Section 15.4.3 of Reference 7 Section 15.4.3 of Reference 8 Section 15.4.3 of Reference 12
(29) UFSAR 15.4.6 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	Pages 8 and 9 of the SE of Reference 2 Sections 15.2.4.6 and 15.4.4 of Reference 7 Section 3.4.7 of Reference 9 Section 15.4.4 of Reference 12
(30) Not in UFSAR Anticipated Transients Without Scram	Section 15.3.6 of Reference 7 Section 15.3.6 of Reference 10

References

1. TVA letter to NRC, CNL-19-082, “License Amendment Request for Measurement Uncertainty Recapture Power Uprate (WBN-TS-19-06),” dated October 10, 2019 (ML19283G119)

2. NRC letter to TVA, “Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Increase of Reactor Power to 3459 Megawatts Thermal (TAC No. MA9152),” dated January 19 2001 (ML010260074)

3. TVA letter to NRC, “Watts Bar Nuclear Plant (WBN) Unit 1 – Updated Final Safety Analysis Report (UFSAR) Amendment 2,” dated April 6, 2001 (ML011020281)
4. TVA letter to NRC, “Watts Bar Nuclear Plant (WBN) – Unit 2 – Reactivation of Construction Activities,” dated August 3, 2007 (ML072190047)
5. TVA letter to NRC, “Watts Bar Nuclear Plant (WBN) – Unit 2 – Regulatory Framework for the Completion of Construction and Licensing Activities for Unit 2,” dated January 29, 2008 (ML080320443)
6. TVA letter to NRC, “Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR), Amendment 97,” dated January 11, 2010 (ML100191421)
7. NUREG-0847, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2 Docket Nos. 50-390 and 50-391,” dated June 1982 (ML072060490)
8. NUREG-0847, Supplement 14, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2 Docket Nos. 50-390 and 50-391,” dated December 1994 (ML072060486)
9. NRC letter to TVA, “Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Revised Technical Specification 4.2.1 ‘Fuel Assemblies’ to Increase the Maximum Number of Tritium Producing Burnable Absorber Rods (CAC No. MF6050),” dated July 29, 2016 (ML16159A057)
10. NUREG-0847, Supplement 24, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2 Docket Nos. 50-390 and 50-391,” dated September 2011 (ML11277A148)
11. NUREG-0847, Supplement 26, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2 Docket Nos. 50-390 and 50-391,” dated June 2013 (ML13205A136)
12. NUREG-0847, Supplement 25, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2 Docket Nos. 50-390 and 50-391,” dated November 2011 (ML12011A024)

NRC RAI NCSG-1:

The guidance in NRC Regulatory Issue Summary (RIS) 2002-03, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,” recommends that a licensee provide information for its flow-accelerated corrosion (FAC) program as part of its license amendment request (LAR). The staff’s acceptance criteria for FAC-related reviews are based on maintaining the minimum acceptable wall thickness for components susceptible to FAC.

Section IV.1.E.iii, “Flow Accelerated Corrosion Program,” of the LAR states that the Watts Bar Unit 2 (WBN2) FAC program is based on the “…latest revision of the Electric Power Research Institute (EPRI) NSAC-202L, ‘Recommendations for an Effective Flow-Accelerated Corrosion Program.’”

In order for the NRC staff to have reasonable assurance the FAC program will continue to manage FAC at the MUR power uprate conditions, the staff needs to ensure the WBN2 licensing basis adequately describes the FAC program. Clarify which revision of NSAC-202L is currently part of the WBN2 licensing basis and provide the basis for using this revision.

TVA Response

As noted in the RAI, Section IV.1.E.iii of Enclosure 2 to the LAR states that the FAC program is based on the latest revision of EPRI NSAC-202L. This statement is also consistent with the TVA FAC program. Revision 4 of EPRI NSAC-202L (Product ID EPRI 3002000563) is the current version. Implementation of the FAC program utilizing the methods contained in this standard are intended to minimize the risk of a FAC-induced failure and to minimize the consequence of FAC-induced wall-thinning.

NRC RAI NCSG-2:

Table IV.1.E-1, "Wear Rate Analysis for Lines with an Expected Increase in Wear Post-MUR Power Uprate," of the LAR provides a wear rate analysis to assess the impacts of the MUR power uprate on certain components susceptible to FAC at WBN2. However, the table appears to provide wear rate values averaged over a given line. While the overall increase in wear rate for a line modeled in CHECWORKS™ may not be significant, individual susceptible components within the line may have significant projected increases in wear rate.
In order to obtain reasonable assurance that components within the lines described by the licensee will not experience significant degradation at the MUR power uprate conditions; the NRC staff requests wear rate values for individual susceptible components in the lines that will experience the greatest increase in wear rate due to the MUR power uprate conditions. Additionally, if any of these components are expected to have significantly increased wear rates, describe how the current FAC program will manage this reduction in component thickness.

TVA Response

The three components in each piping section in the WBN2 CHECWORKS SFA Model with the greatest predicted increase (i.e., percent change) in wear rate are provided in Table IV.1.E-2.

No significant increases in wear rates were identified for MUR power uprate conditions. The majority of components show a predicted wear rate increase of approximately 3% or less. There are no components identified where the predicted increase in FAC wear rates are greater than 5%. Therefore, the increases in wear rates due to the MUR power uprate are considered minor and the existing FAC Program is adequate to incorporate the updated predictions and manage the potential reduction in component thickness.

Table IV.1.E-2: Wear Rate Analysis for Components with the Greatest Predicted Percent Increase in Wear Post-MUR Power Uprate

Piping section in WBN Unit 2 CHECWORKS SFA Model	Component	Maximum Percent Increase in Wear Rate Post-Uprate	Maximum Wear Rate Post-Uprate (mils/yr)
CD COND BP TO FWH4	202BC148P	1.8%	1.058
CD COND BP TO FWH4	202BC153P	1.8%	1.058
CD COND BP TO FWH4	202BC158P	1.8%	1.058
CD FWH 4 to FWH 3	202CC006P	1.5%	0.987
CD FWH 4 to FWH 3	202CC015P	1.5%	0.987
CD FWH 4 to FWH 3	202CC029P	1.5%	0.987
CD FWH3 TO FWH2	202DC028T	1.3%	1.828
CD FWH3 TO FWH2	202DC044T	1.3%	1.828
CD FWH3 TO FWH2	202DC011P	1.2%	1.966

Table IV.1.E-2: Wear Rate Analysis for Components with the Greatest Predicted Percent Increase in Wear Post-MUR Power Uprate			
Piping section in WBN Unit 2 CHECWORKS SFA Model	**Component**	**Maximum Percent Increase in Wear Rate Post-Uprate**	**Maximum Wear Rate Post-Uprate (mils/yr)**
CD FWH5 to COND BP	202BC009P	1.6%	0.808
CD FWH5 to COND BP	202BC058R	1.4%	2.175
CD FWH5 to COND BP	202BC010X	1.4%	2.086
CD FWH6 TO FWH5	202AC005P	1.2%	1.305
CD FWH6 TO FWH5	202AC007P	1.2%	1.305
CD FWH6 TO FWH5	202AC009P	1.2%	1.305
CD SG HT EX TO CBP	202BC048P	0.8%	1.316
CD SG HT EX TO CBP	202BC053T	0.8%	2.802
CD SG HT EX TO CBP	202BC044E	0.8%	2.203
ES HP Turb to FWH1	2051C087P	3.2%	4.343
ES HP Turb to FWH1	2051C002P	2.9%	6.202
ES HP Turb to FWH1	2051C064P	2.9%	6.202
ES HP Turb to FWH2	2052C007P	3.5%	7.553
ES HP Turb to FWH2	2052C028P	3.1%	8.421
ES HP Turb to FWH2	2052C030P	2.6%	7.765
ES HP Turb to FWH3	2053C044N	1.3%	21.372
ES HP Turb to FWH3	2053C051N	1.3%	21.372
ES HP Turb to FWH3	2053C058N	1.3%	21.372
ES HP Turb to MSR1	2051C132P	3.1%	2.269
ES HP Turb to MSR1	2051C135aP	3.0%	2.669
ES HP Turb to MSR1	2051C148P	3.0%	2.669
ES HP Turb to MSR2	2051C017P	3.1%	2.269
ES HP Turb to MSR2	2051C025aP	3.0%	2.669
ES HP Turb to MSR2	2051C037P	3.0%	2.669
ES LP Turb to FWH6	2056C005N	3.1%	9.278
ES LP Turb to FWH6	2056C010N	3.1%	9.278
ES LP Turb to FWH6	2056C015N	3.1%	9.278
FW FWH1 TO SG	203BC327P	3.0%	0.490
FW FWH1 TO SG	203BC330P	3.0%	0.490
FW FWH1 TO SG	203BC417P	3.0%	0.490
HD FWH 1 to FWH 2	206DC043P	2.5%	0.299
HD FWH 1 to FWH 2	206DC041E	2.5%	0.711
HD FWH 1 to FWH 2	206DC040P	2.5%	0.490
HD FWH 2 to No.3 HDT	206FC001N	2.0%	2.029
HD FWH 2 to No.3 HDT	206FC023N	2.0%	2.029
HD FWH 2 to No.3 HDT	206FC045N	2.0%	2.029
HD FWH 3 to No.3 HDT	206FC089P	0.3%	0.213
HD FWH 3 to No.3 HDT	206FC102P	0.3%	0.294
HD FWH 3 to No.3 HDT	206FC103E	0.3%	0.350
HD FWH 4 to FWH 5	206GC038P	2.5%	0.579

Table IV.1.E-2: Wear Rate Analysis for Components with the Greatest Predicted Percent Increase in Wear Post-MUR Power Uprate			
Piping section in WBN Unit 2 CHECWORKS SFA Model	**Component**	**Maximum Percent Increase in Wear Rate Post-Uprate**	**Maximum Wear Rate Post-Uprate (mils/yr)**
HD FWH 4 to FWH 5	206GC039E	2.5%	1.329
HD FWH 4 to FWH 5	206GC040N	2.5%	1.487
HD FWH 5 to FWH 6	206GC139N	5.0%	1.304
HD FWH 5 to FWH 6	206GC158N	5.0%	1.304
HD FWH 5 to FWH 6	206GC177N	5.0%	1.304
HD LP RHR to FWH2	206BC055P	3.4%	0.197
HD LP RHR to FWH2	206BC092P	3.4%	0.197
HD LP RHR to FWH2	206BC132P	3.4%	0.197
HD MSR to No.3 HDTank	206AC163P	2.2%	0.449
HD MSR to No.3 HDTank	206AC185P	2.2%	0.449
HD MSR to No.3 HDTank	206AC218P	2.2%	0.449
HD No3 HD Tnk to Cond	206FC154P	1.7%	0.914
HD No3 HD Tnk to Cond	206FC116aP	1.7%	0.563
HD No3 HD Tnk to Cond	206FC116bP	1.7%	0.563

NRC RAI NCSG-3:

SRP Section 6.1.2, “Protective Coating Systems (Paints) – Organic Materials,” Revision 3 (ADAMS Accession No. ML063600399), provides the NRC staff guidance to ensure coating systems used inside containment are evaluated to determine suitability for design basis accident (DBA) conditions. This guidance directs the reviewer to verify coating monitoring and maintenance procedures are capable of ensuring that coatings will not fail and become a debris source for the emergency core cooling system. This guidance also instructs the reviewer to determine the suitability of the protective coatings in the DBA environment when exposed to high temperatures, pressures, and radiation dose.

Section VII.6.B, “Containment Coatings Program,” of the LAR discusses the current licensing basis for the WBN2 containment coatings program as well as the DBA qualifications of the coatings in containment. This section of the LAR references Updated Final Safety Analysis Report (UFSAR) Section 6.1.4, “Degree of Compliance with Regulatory Guide [RG] 1.54 for Paints and Coatings Inside Containment,” for a description of the coatings program basis. However, the staff requests the following clarification on both the licensing basis as well as the DBA qualifications for coatings in containment:

a. UFSAR Section 6.1.4 states that WBN2 follows RG 1.54 except for the endorsement of American National Standards Institute (ANSI) N101.4, “Quality Assurance for Protective Coatings Applied to Nuclear Facilities,” in paragraph C.1 (ADAMS Accession No. ML19336A067). The UFSAR also states that applicable provisions in ANSI N101.4 are incorporated into the coatings program. Confirm that the basis for the WBN2 coatings program in NUREG-0847, “Safety Evaluation Report [SER] Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2,” as well as Supplement 22 to the SER, still apply to the WBN2 current licensing basis.

b. *For the staff to verify that the qualifications of containment coatings are still bounding for the proposed MUR DBA conditions, provide a comparison of DBA conditions (e.g., temperature, pressure, dose) to the qualification conditions for the containment coatings.*

<u>TVA Response</u>

a. In Section 6.1.2, "Organic Materials," of NUREG-0847 (Reference 1), NRC evaluated the TVA protective coatings program and stated, "The staff concludes that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR 50. This conclusion is based on the coating systems and their applications meeting (1) the positions of Regulatory Guide (RG) 1.54, with an acceptable alternate to ANSI N101.4 (1972) and (2) the testing requirements of ANSI N101.2."

   In Section 6.1.2, "Organic Materials," of Supplement 22 to NUREG-0847 (Reference 2), NRC further stated that TVA has "maintained its commitment to meet the positions of RG 1.54, with the acceptable alternative to ANSI N101.4-1972 and the testing requirements of ANSI N101.2-1972."

   Additional information on the TVA protective coatings program for WBN is also provided in Enclosure 3 to the TVA response to Generic Letter 98-04 (Reference 3).

   The information in WBN UFSAR Section 6.1.4 is consistent with the information in the above referenced documents; therefore, TVA confirms that the basis for the WBN2 coatings program in NUREG-0847, and Supplement 22 to the SER, still apply to the WBN2 current licensing basis.

   <u>References</u>

   1. NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2 Docket Nos. 50-390 and 50-391," dated June 1982 (ML072060490)
   2. NUREG-0847, Supplement 22, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2 Docket Nos. 50-390 and 50-391," dated February 2011 (ML110390197)
   3. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN), and Watts Bar Nuclear Plant (WBN), 120-Day Response Generic Letter (GL) 98-04, 'Potential for Degradation of the Emergency Core Colling System (ECCS) and the Containment Spray System (CSS) after a Loss-of-Coolant Accident (LOCA) because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,' dated July 14, 1998," dated November 10, 1998 (ML082460076)

b. Service Level (CSL) I coatings are qualified coatings inside the reactor containment where coating failure could adversely affect the operation of post-accident fluid systems and, thereby, impair safe shutdown. Section VII.6.B of the LAR indicates that the DBA pressure/temperature profiles and dose analyses are not impacted and remain applicable for MUR power uprate. Therefore, the requirements for CSL I coatings inside containment are not impacted.

   The CSL I coating systems applied at WBN Unit 2 include Keeler and Long (KL) 6129/5000, KL 4129/4500, Carbozinc 11 SG, and Amerlock 400 NT. To demonstrate that the test qualification conditions for CSL I coatings applied at WBN Unit 2 bound the analyzed DBA conditions for the current licensing basis, the coating systems qualification information was reviewed. The maximum coating qualification temperature (for all CSL I coatings) is 340°F and bounds the peak DBA temperature of 327°F. The peak DBA pressure, 9.36 psig, is bounded by the CSL I coating test pressures, which varied during testing ranging from up to 70 psig initially to a minimum of approximately 10 psig. The test condition for integrated radiation dose is 1E9 rads for all CSL I coatings which bounds the maximum DBA integrated radiation dose (approximately 3E8 rads). The CSL I coating test conditions (temperature, pressure, and dose) bound the DBA profile peak values, which bound the MUR power uprate conditions. Therefore, the existing CSL I coating qualification basis remains valid and the qualified coatings are acceptable for MUR power uprate conditions.

NRC RAI NCSG-4

The bases for WBN2 Technical Specification 3.4.17, “SG Tube Integrity” (ADAMS Accession No. ML13357A054), state that the basis for the WBN2 Steam Generator Program is Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” and its referenced EPRI Guidelines. These referenced EPRI Guidelines include the EPRI “Pressurized Water Reactor [PWR] Primary Water Chemistry Guidelines.” The EPRI PWR Primary Water Chemistry Guidelines contain limits on specific impurities for primary water chemistry and associated actions if these impurity limits are not met.

In order to ensure the integrity of the steam generator (SG) tubes can be maintained at MUR power uprate conditions, the NRC staff reviewed the primary water chemistry program. UFSAR Table 5.2-10, “Reactor Coolant Water Chemistry Specifications,” provides a maximum concentration of chlorides and fluorides of 0.15 parts per million (ppm) and states that the concentration of oxygen will be maintained below 0.1 ppm (ADAMS Accession No. ML19176A139). These values are greater than EPRI Primary Water Chemistry Guidelines, Revision 7, action level 1 limits for primary water chemistry parameters and may contribute to degradation of Alloy 600 SG tubes. Provide the justification for why operations at the MUR power uprate conditions will be able to maintain SG tube integrity with the primary water chemistry limits described in the WBN2 UFSAR.

TVA Response

Section 3.5 of the EPRI PWR Primary Water Chemistry Guidelines defines three action levels as follows.

> “The chemistry control parameters, action levels values, hold limits and monitoring frequencies presented herein are appropriate for protecting primary system pressure boundary integrity and/or fuel integrity. Three action levels have been defined for remedial actions to be taken when parameters are outside the control values. Corrective actions in response to exceeding an Action Level must in all cases be consistent with plant Technical Specifications / Technical Requirements Manual.”
>
> “3.5.2.1 Action Level 1
> The Action Level 1 value of a parameter represents the threshold value, beyond which plant or research data or engineering judgment indicates that long-term system reliability may be affected, thereby warranting an improvement of operating practices.”

"3.5.2.2 Action Level 2
The Action Level 2 value of a parameter represents the threshold value, beyond which plant or research data or engineering judgment indicates significant damage could be done to the system in the short term, thereby warranting a prompt correction of the abnormal condition."

"3.5.2.3 Action Level 3
The Action Level 3 value of a parameter is the threshold value, beyond which plant or research data or engineering judgment indicates that it is inadvisable to continue to operate the plant."

The primary water chemistry limits described in WBN Unit 2 UFSAR Table 5.2-10 are based on the action 2 level limits from the EPRI Primary Water Chemistry Guidelines. These limits are also consistent with the WBN Unit 2 Technical Requirements Manual Section 3.4.4, "Reactor Coolant System (RCS)." The chemistry limits in WBN Unit 2 UFSAR Table 5.2-10 are also consistent with other utilities FSARs (e.g., ML19360A116 for the Salem Generating Station Units 1 and 2, ML19296C741 for the Vogtle Electric Generating Plants, Units 1 and 2).

As noted in Section 3.5.2 of the EPRI Primary Water Chemistry Guidelines:

"Actions to be taken if a parameter exceeds the Action Level 1 value:

- Efforts shall be made to bring the parameter to below the Action Level 1 value within seven days.
- If the parameter has not been restored to below the Action Level 1 value within seven days, a technical review* shall be performed and a program for implementing corrective measures instituted. Such a program may require equipment additions or modifications over the long term.

    * It is required that each plant perform a formal technical review for prolonged abnormal water chemistry conditions. The review shall address an evaluation of the condition, informing appropriate personnel (e.g., those responsible for fuel integrity, primary system integrity and radiation management) and levels of management of the existence of the condition and its implications, and the possible corrective measures over the short and long terms."

"Actions to be taken if a parameter exceeds the Action Level 2 value:

- Efforts shall be made to bring the parameter to below the Action Level 2 value within 24 hours.
- If the parameter has not been restored to below the Action Level 2 value within 24 hours, an orderly shutdown shall be initiated and the plant shall be brought to a coolant temperature < 250°F (121°C) as quickly as safe plant operation permits. If chemistry is improved to below the Action Level 2 value prior to plant shutdown, full power operation may be resumed.
- Following an Action Level 2 event, a technical review* of the incident shall be performed and appropriate corrective measures taken before the unit is restarted.

    * It is required that each plant perform a formal technical review for prolonged abnormal water chemistry conditions. The review shall address an evaluation of the condition, informing appropriate personnel (e.g., those responsible for fuel integrity, primary system integrity and radiation management) and levels of

management of the existence of the condition and its implications, and the possible corrective measures over the short and long terms."

"Actions to be taken if a parameter exceeds the Action Level 3 value:

- An orderly unit shutdown shall be initiated immediately, with reduction of coolant temperature to < 250°F (121°C) as quickly as safe plant operation permits.*

- Following an Action Level 3 event, a technical review** of the incident shall be performed and appropriate corrective measures taken before the unit is restarted.

  * If chemistry is improved to within the requirements of Action Level 3 prior to plant shutdown, power operation may be resumed, subject to the requirements of other Action Levels and plant Technical Specifications.

  ** It is required that each plant perform a formal technical review for prolonged abnormal water chemistry conditions. The review shall address an evaluation of the condition, informing appropriate personnel (e.g., those responsible for fuel integrity, primary system integrity and radiation management) and levels of management of the existence of the condition and its implications, and the possible corrective measures over the short and long terms."

The WBN procedure for System Chemistry Specifications provides parameters, monitoring frequencies, specifications, and corrective actions used to evaluate chemistry conditions in plant systems. This procedure reflects the guidance presented in the EPRI PWR Secondary Water Chemistry Guidelines, PWR Primary Water Chemistry Guidelines, PWR Primary to Secondary Leak Guidelines, and Closed Cooling Water Chemistry Guidelines. This procedure adheres to the action levels 1, 2, and 3 with the same actions to be taken as in the EPRI guidelines.

Adhering to the EPRI PWR Primary Water Chemistry Guidelines, as well as the other guidelines noted in the above paragraph, provide assurance that operations at the MUR power uprate conditions will be able to maintain SG tube integrity with the primary water chemistry limits described in the WBN2 UFSAR.

<u>NRC EENB RAIs</u>

<u>Regulatory Criteria:</u>

Paragraph 50.49(e)(1) of 10 CFR requires that the time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during and following which this equipment is required to remain functional.

Paragraph 50.49(b)(2) 10 CFR requires qualification of nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (b)(1)(i)(A) through (C) of paragraph (b)(1) of 10 CFR 50.49 by the safety-related equipment.

<u>Issue</u>:

In the LAR, TVA noted that they have evaluated the impact of the proposed MUR power uprate on the Environmental Qualification (EQ) of equipment. TVA asserted that the results of their evaluations showed that electrical equipment that is required to be environmentally qualified per

10 CFR 50.49 will remain qualified (i.e., bounded by the existing EQ). However, TVA did not provide enough detail for the staff to confirm TVA's conclusion.

It is also unclear as to whether TVA considered the impact of the proposed change on qualified non-safety related equipment (under 10 CFR 50.49(b)(2)) whose failure under postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment.

NRC RAI EENB-1:

In the LAR, TVA stated that:

> *The evaluation of the systems inside containment and in the MSVV for accident temperature and pressure conditions showed that the current design basis analyses were performed at 102% of 3411 MWt (i.e., 3479 MWt), which bounds the MUR power uprate. There is no EQ impact with respect to temperature or pressure due to the MUR power uprate. No areas transition from mild to harsh environments because of the MUR power uprate based on temperatures.*

Based on various statements in the LAR, it's unclear to the NRC staff as to whether the existing accident analyses for all areas of the plant were performed at 102% rated thermal power (RTP) versus being limited to inside containment and the main steam valve vault. If the accident analyses performed at 102% RTP were limited to inside containment and the main steam valve vault, provide an evaluation that shows that the environmental qualification remains bounding for electric equipment located in areas of the plant that will experience parameter changes (i.e., increase in temperature, pressure, radiation, humidity, chemical spray, etc.) due to the proposed MUR power uprate.

TVA Response

The quoted text from the LAR was not intended to imply that the design basis accident analyses for the plant were not performed at 102% RTP or that the accident analyses were limited to the containment and main steam valve vault. The determination of locations in the plant that are classified as harsh environments and the associated environmental conditions are based on the design basis accident analyses (e.g., LOCA, high energy line break) or more conservative bounding analyses that are not affected by MUR power uprate conditions. For example, some line break analyses are based on hot zero power conditions and some dose analyses are based on an assumed power level significantly greater than 102% RTP. The accident analyses assume 102% RTP regardless of location in the plant. The requirements for EQ of safety-related electrical equipment located inside harsh environments are selected to bound the environmental conditions associated with the accident and post-accident environment. Temperature, relative humidity, pressure, radiation dose, area type, chemical spray, and flooding were the parameters considered for qualification of electrical equipment.

Areas of the plant with both mild and harsh environments were reviewed for impacts associated with MUR power uprate. No areas transition from mild to harsh environments due to the proposed uprate and no impacts were identified for the EQ Program. The conditions used in the EQ Program are based on accident analyses that bound the MUR power uprate thermal power level. These conditions are applied to all areas of the plant and are not limited to inside containment and the main steam valve vault. Therefore, the conditions defined in the existing EQ program remain acceptable for MUR power uprate.

NRC RAI EENB-2:

In Enclosure 2 – V Electrical Equipment Design – V.1.C, “EQ of Electrical Equipment,” TVA stated that:

> *The TVA EQ Program addresses safety-related electrical equipment within the scope of 10 CFR 50.49 for WBN. The EQ program for WBN was reviewed to evaluate the impact of the MUR power uprate and it was determined that no programmatic changes are required. See Section II.1.D.iii (Item 32).*

Explain how TVA has assessed the impact of the proposed change on qualified non-safety related equipment (under 10 CFR 50.49(b)(2)) whose failure in postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment.

<u>TVA Response</u>

The analyses of record which document compliance with 10 CFR 50.49(b)(2) for non-safety related electrical equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions performed by safety-related equipment were reviewed and determined to not be impacted by MUR power uprate conditions.

The analyses encompass, but were not limited to, electrical equipment whose failure is subject to postulated harsh environmental conditions. Analysis was performed for non-safety related protective devices and the related associated circuits. The analysis indicated that none of the identified protective devices were subjected to a harsh environment when the device may be required to function. Additional analysis was performed for the non-safety interfaces, regardless of whether the non-safety system component locations or cable routing were subject to harsh environments. The analysis concluded that no environmentally-induced failure of non-safety related electric equipment was found which could prevent accomplishment of required safety functions. The conclusions of these analyses are not affected by WBN MUR power uprate.

NRC RAI EMIB-1

The WBN2 LAR does not provide any evaluation of snubbers (similar to pumps and valves) in the submitted WBN2 MUR power uprate application. Please describe the snubber evaluation and its results. If an evaluation was not performed, justify that the existing evaluation of the snubbers is bounding for the uprated power.

<u>TVA Response</u>

The TVA In-Service Testing (IST) Program includes snubbers (dynamic restraints) that are required to ensure the integrity of the reactor coolant pressure boundary; or required for systems and components that perform a specific function to bring the reactor to a safe shutdown condition, maintain the safe shutdown condition, or mitigate the consequences of an accident.

The systems within the scope of the IST Program were reviewed and determined not to be impacted by MUR power uprate. No changes were identified for any of the associated piping analyses. Therefore, support loads are not changed and the snubbers are not affected by the MUR power uprate.

NRC RAI EMIB-2

The WBN2 measurement uncertainty recapture (MUR) LAR, Table IV.1-1, shows that for analyzed MUR power uprate Cases 1 and 2 the total steam outlet flow is increased. The WBN2 MUR LAR, Sections, IV.1.B.iii, "Flow Induced Vibration," and IV.1.F, related to Steam Generator Tubing, do not discuss any evaluation of safety-related components in the steam system due to increased steam flow. Please explain how the adverse effects from flow-induced vibration of safety-related components in the steam system were evaluated due to increased steam flow.

TVA Response

Flow induced vibration (FIV) monitoring of the main steam system was performed during WBN Unit 2 initial plant startup power ascension testing up to 100% power. No excessive FIV of any safety-related components in the main steam system were identified because of that testing. Since completion of initial plant startup testing, no excessive flow induced vibrations of any safety-related components in the main steam system have been identified.

Flow velocities in the main steam system piping will increase less than 2% because of the MUR power uprate. FIV in the main steam system are expected to increase approximately proportional to the increase in flow velocity squared, or less than 4%. WBN Unit 1 has been operating at MUR power uprate flow velocities and has not experienced any excessive FIV of safety related components in the main steam system.

Based on the WBN Unit 2 initial startup test results, the WBN Unit 2 operating experience since initial startup, the small increase in FIV amplitudes expected as a result of the MUR power uprate, and the WBN Unit 1 operating experience at MUR power uprate flow rates, no adverse effects from FIV of safety-related components in the main steam system are anticipated as result of the WBN2 MUR power uprate.

Revised Description of the Proposed Change

2.0 DETAILED DESCRIPTION

2.1 DESCRIPTION OF THE PROPOSED CHANGE

The following changes are being made by this LAR.

1. WBN Unit 2 OL Item 2.C.(1) is being revised to increase the maximum core power level from 3411 MWt to 3459 MWt.

2. The definition of RTP in TS 1.1, “Definitions,” is being changed to account for the increase in reactor core thermal power level as follows:

   “RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.”

3. TS 5.9.5b, “CORE OPERATING LIMITS REPORT (COLR),” currently states:

   “The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:”

   TS 5.9.5b is being revised as follows:

   “The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% RTP is specified in a previously approved method, 100.6% RTP may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 11 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102% RTP (3411 MWt) shall be used. The approved analytical methods are specifically those described in the following documents”:

4. The NRC approved Caldon Topical Reports for LEFMs are being added as document number 11 in the list of documents in TS 5.9.5b as follows:

   “11. Caldon, Inc., Engineering Report-80P, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System,” Revision 0; and Caldon Ultrasonics Engineering Report ER-157P-A, “Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or LEFM CheckPlus System," Revision 8 and Revision 8 errata.”

Enclosure 3

Revised Proposed WBN Unit 2 TS 5.9.5.b (Markup)

5.9 Reporting Requirements (continued)

5.9.3 Radioactive Effluent Release Report

---NOTE---
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.9.4 Reserved for Future Use

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to the initial and each reload cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

LCO 3.1.4	Moderator Temperature Coefficient
LCO 3.1.6	Shutdown Bank Insertion Limits
LCO 3.1.7	Control Bank Insertion Limits
LCO 3.2.1	Heat Flux Hot Channel Factor
LCO 3.2.2	Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3	Axial Flux Difference
LCO 3.9.1	Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% RTP is specified in a previously approved method, 100.6% RTP may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 11 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102% RTP (3411 MWt) shall be used. The approved analytical methods are, specifically those described in the following documents:

(continued)

Enclosure 4

Revised Proposed WBN Unit 2 TS 5.9.5.b (Re-typed)

5.9 Reporting Requirements (continued)

5.9.3 Radioactive Effluent Release Report

---NOTE---
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.9.4 Reserved for Future Use

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to the initial and each reload cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

LCO 3.1.4	Moderator Temperature Coefficient
LCO 3.1.6	Shutdown Bank Insertion Limits
LCO 3.1.7	Control Bank Insertion Limits
LCO 3.2.1	Heat Flux Hot Channel Factor
LCO 3.2.2	Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3	Axial Flux Difference
LCO 3.9.1	Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% RTP is specified in a previously approved method, 100.6% RTP may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 11 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102% RTP (3411 MWt) shall be used. The approved analytical methods are, specifically those described in the following documents:

(continued)

Enclosure 5

Proposed Update to Table 15.1-2 of the WBN dual-unit UFSAR
(For Information Only)

WBN

TABLE 15.1-1

NUCLEAR STEAM SUPPLY POWER RATINGS AND FLOWRATES

	UNIT 1	UNIT 2
Guaranteed Nuclear Steam Supply System thermal power output	3425 MWt[1]	3427 MWt (1)
The Engineered Safety (Features) Design Rating (ESDR)(initial design maximum calculated turbine rating is 3579 MWt)	3650 MWt	3650 MWt
Thermal power generated primarily by the reactor coolant pumps	15.21 MWt[1]	16 MWt (1)
Guaranteed core thermal power	3411 MWt[1]	3411 MWt (1)
RCS Thermal Design Flow	372400 gpm	~~–~~ 372400 gpm
RCS Minimum Measured Flow	379100 gpm	~~–~~ 379100 gpm

NOTE:

1. The safety analyses completed for Watts Bar also support an uprated core thermal power level of 3459 MWt and a NSSS power of ~~3474.21~~ 3475 MWt (~~using the Watts Bar specific NHI value of 15.21 MWt~~ which bounds the Unit 1 and Unit 2 specific NHI values of 15.21 MWt and 16.0 MWt, respectively), based on a redefinition of the 2% power uncertainty (2% to 0.6%).

TABLE 15.1-2 (Sheet 1 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED FOR: MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED ~~[1, 5,6]~~ [1,6] (MWt)
CONDITION II					
Uncontrolled RCC Assembly Bank Withdrawal from Subcritical Condition	TWINKLE, FACTRAN, VIPRE-01	Refer to Section 15.2.1.2 (Part 2)	--	Least negative Doppler power coefficient-Doppler defect = 960 pcm	~~3411~~ 3459[5] (critical @ 0.0 fraction of Nominal [FON])
Uncontrolled RCC Assembly Bank Withdrawal at Power	LOFTRAN	---	0.0 and 0.43	lower and upper[2]	~~3425~~ 3475
RCC Assembly Misalignment	VIPRE-01, LOFTRAN	---	0.0	upper[2]	~~3425~~ 3459[5]
Uncontrolled Boron Dilution	NA	NA	NA	NA	0 and ~~3425~~ 3475
Partial Loss of Forced Reactor Coolant Flow	LOFTRAN, VIPRE-01, FACTRAN	---	0.0	upper[2]	3475
Startup of an Inactive Reactor Coolant Loop	NA	---	NA	NA	NA
Loss of External Electrical Load and/or Turbine Trip	LOFTRAN	---	0.0	upper[2]	3475
Loss of Normal Feedwater/ Loss of Offsite Power to the Station Auxiliaries	LOFTRAN	--	0.0	upper[2]	3475
Excessive Heat Removal Due to Feedwater System Malfunctions	LOFTRAN	---	0.43	lower[2]	3475 ~~(Unit 1)~~ ~~3425 (Unit 2)~~

TABLE 15.1-2 (Sheet 2 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED FOR: MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED ~~[1, 5,6]~~ [1,6] (MWt)
Excessive Load Increase Incident	NA	---	NA	NA	~~NA~~ 3475
Accidental Depressurization of the Reactor Coolant System	LOFTRAN	---	0.0	upper[2]	~~3425~~ 3475
Accidental Depressurization of the Main Steam System	Accident evaluated; bounded by major rupture of a steam pipe				
Inadvertent Operation of ECCS During Power Operation	LOFTRAN	---	0.0 and 0.43	lower and upper[2]	3475 ~~(Unit 1)~~ ~~3475[7] (Unit 2)~~
CONDITION III					
Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates Emergency Core Cooling	NOTRUMP, LOCTA-IV				~~3411[4] (Unit 1)~~ 3459[4,5] ~~3411[7] (Unit 2)~~
Inadvertent Loading of a Fuel Assembly into an Improper Position	LEOPARD, TURTLE	---	Minimum	NA	~~3425~~ 3475
Complete Loss of Forced Reactor Coolant Flow	VIPRE-01, FACTRAN, LOFTRAN	---	0.0	upper[2]	~~3425 (Unit 1)~~ 3475 ~~(Unit 2)~~
Waste Gas Decay Tank	NA	---	NA	NA	~~3579~~ 3475
Single RCC Assembly Withdrawal at Full Power	TURTLE, VIPRE-01, LEOPARD	---	NA	NA	~~3425~~ 3459[5]
CVCS Malfunction During Power Operation	LOFTRAN	---	0.43	lower[2]	3475 (Unit 2)

TABLE 15.1-2 (Sheet 3 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED FOR: MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED~~[1, 5,6]~~ [1,6] (MWt)
CONDITION IV					
Major Rupture of Pipes Containing Reactor Coolant Up to and Including Double-ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss of Coolant Accident)	SATAN-VI, WREFLOOD, LOTIC 2, FROTH, WCOBRA/TRAC, MONTECF, HOTSPOT, RSURF (Unit 1), WCOBRA\TRAC, HOTSPOT, LOTIC2 (Unit 2)	---	0	Function of fuel temperature.	~~3459[4] (Unit 1)~~ 3459[4,5] ~~3475 (Unit 2)~~
Major Rupture of a Steam Pipe	LOFTRAN, VIPRE-01	Function of moderator density; see Section 15.2.13 (Figure 15.2-40)		Note 3	3475 ~~(Unit 1)~~ ~~3425 (Unit 2)~~ (critical @ 0.0 fraction of nominal [FON]).
Major Rupture of a Main Feedwater Pipe	LOFTRAN	---	0.0	lower[2]	3475 ~~(Unit 1)~~ ~~3425 (Unit 2)~~
Steam Generator Tube Rupture	LOFTTR2	0 pcm/°F @ 100 RTP	Figure 15.1-7 (Unit 2)	upper[2]	~~3425 (Unit 1)~~ 3475 ~~3427 (Unit 2)~~
Single Reactor Coolant Pump Locked Rotor	LOFTRAN, VIPRE-01 FACTRAN	---	0.0	upper[2]	3475
Fuel Handling Accident	NA	NA	NA		~~3579~~ 3480[5] (Unit 1) 3565[5] (Unit 2)

TABLE 15.1-2 (Sheet 4 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED FOR: MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED~~[1, 5,6]~~ [1,6] (MWt)
Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)	TWINKLE, FACTRAN	Refer to Section 15.4.6	---	Least negative Doppler defect; see Table 15.4-12	~~3411 (HZP 0)~~ 0 and 3459[5]

[1] The values provided do not include the power uncertainty that is applied either directly (non-RTDP events) or statistically (RTDP events).
[2] Refer to Figure 15.1-5.
[3] Refer to Figure 15.4-9.
[4] LOCA M/E based on Engineering Safety Design Rating (ESDR) of 3650 MWt.
~~[5] The 14 MWt value is based on a generic calculation for a representative 4-loop design. The Watts Bar specific value is 16.0 MWt. Thus the actual NSSS thermal output can be as high as 3475 MWt with a licensed core power of 3459 MWt.~~
~~[6] Although several of these analyses are based upon a core power of 3411 MWt and NSSS power of 3425 MWt, an uprated core power of 3459 MWt and NSSS power of 3475 MWt are also supported via evaluation, based upon a redefinition of the 2% power uncertainty (2% to 0.6%). However, the Unit 1 NSSS will operate at a maximum power value of 3,474.21 MWt based on a revised NHI value of 15.21 (previously 16.0) MWt. Therefore, the previous NSSS thermal power output of 3,475 MWt assumed in the analyses remains bounding. (Unit 1)~~
~~[7] Several of these analyses are conservatively based upon a core power of 3459 MWt and NSSS power of 3475 MWt, based upon a redefinition of the 2% power uncertainty (2% to 0.6%), which bounds a core power of 3411 MWt and NSSS power of 3425 MWt. (Unit 2)~~

[5] These analyses are based on core power instead of NSSS power.

[6] These analyses support an uprated core power of 3459 MWt and NSSS power of 3475 MWt via explicit analysis or evaluation based upon a core power of 3411 MWt and NSSS power of 3425 MWt and a redefinition of the 2% power uncertainty (2% to 0.6%).