1101 Market Street, Chattanooga, Tennessee 37402

CNL-21-010

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

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

Subject: **Correction of Application to Implement the FULL SPECTRUM™[1] LOCA (FSLOCA™[1]) Methodology for Loss-of-Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04) (EPID L-2020-LLA-0005)**

References:

1. TVA Letter to NRC, CNL-19-051, "Application to Implement the FULL SPECTRUM™[1] LOCA (FSLOCA™[1]) Methodology for Loss-of-Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN TS 19-04)," dated January 17, 2020 (ML20017A338)

2. TVA letter to NRC, CNL-20-061, "Response to NRC Request for Additional Information Regarding Application to Implement the FULL SPECTRUM™1 LOCA (FSLOCA™1) Methodology for Loss-of-Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN TS 19-04) (EPID L-2020-LLA-0005)," dated August 27, 2020 (ML20240A324)

In Reference 1, Tennessee Valley Authority (TVA) submitted a request for amendments to Facility Operating License (OL) Nos. NPF-90 and NPF-96 for the Watts Bar Nuclear Plant (WBN), Units 1 and 2, respectively. This license amendment request (LAR) revises the WBN Units 1 and 2 Technical Specification (TS) 5.9.5, "Core Operating Limits Report," to replace the loss-of-coolant accident (LOCA) analysis evaluation model references with reference to the FULL SPECTRUM™ Loss-of-Coolant Accident (FSLOCA™) Evaluation Model analysis

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applicable to WBN Units 1 and 2, with replacement steam generators. The proposed change also revises the WBN Unit 2 Operating License (OL) condition 2.C(4) to reflect the implementation of the FSLOCA Evaluation Model methodology. The proposed change also revises WBN Unit 1 TS 4.2.1, "Fuel Assemblies," to delete discussion of Zircalloy fuel rods. In Reference 2, TVA responded to a Nuclear Regulatory Commission (NRC) request for additional information (RAI).

NRC and the Westinghouse Electric Company LLC (Westinghouse) have identified information in Enclosures 1 and 2 to Reference 1 that Westinghouse considers to be proprietary in nature, which was not previously marked as proprietary in Enclosure 1 to Reference 1 and, therefore, not redacted in Enclosure 2 to Reference 1. Accordingly, TVA is resubmitting the information in Reference 1, to properly identify and redact the information that Westinghouse considers to be proprietary in nature. Enclosure 1 to this letter provides the proprietary version of the Evaluation of the Proposed Change including the description and assessment of the proposed change, regulatory analyses, and environmental considerations. The information that was not previously marked as Westinghouse proprietary is highlighted on pages E1-6 and E1-7 of Enclosure 1 and appropriately redacted in Enclosure 2. Additionally, TVA has corrected the WBN Unit 1 Facility Operating License No. in Section 5.3 of Enclosures 1 and 2 from NFP-90 to NPF-90. There are no other changes to the information provided in Enclosures 1 and 2 and their attachments. TVA has entered the error regarding the proprietary markings into the TVA corrective action program.

Enclosure 1 contains information that Westinghouse Electric Company LLC considers to be proprietary in nature and subsequently, pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4). TVA requests that the proprietary information be withheld from public disclosure. Enclosure 2 to this letter provides the non-proprietary version of the description and assessment of the proposed change.

Enclosure 3 to this letter provides a summary of the WBN Units 1 and 2 LOCA Analysis with the FSLOCA Methodology. This document contains information that Westinghouse Electric Company LLC considers to be proprietary in nature and subsequently, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4). TVA requests that this proprietary information be withheld from public disclosure. Enclosure 4 contains a non-proprietary version of the summary of the WBN Units 1 and 2 LOCA Analysis with the FSLOCA Methodology. Enclosures 3 and 4 are unchanged from Reference 1.

Enclosure 5 provides the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-21-5138 affidavit supporting the proprietary withholding requests. It is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, TVA requests that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavits should reference CAW-21-5138 and should be addressed to Camille T. Zozula, Manager, Regulatory Compliance and Corporate Licensing, Westinghouse Electric Company, 1000 Westinghouse Drive, Suite 165, Cranberry Township, Pennsylvania 16066. The CAW-21-5138 affidavit replaces the two affidavits in Enclosure 5 to Reference 1.

This letter does not change the no significant hazard considerations or the environmental considerations contained in Reference 1. This letter also does not impact the information provided in Reference 2. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the enclosure 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 Manager, Fleet Licensing, at (423) 751-3275.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 26th day of January 2021.

Respectfully,

James T. Polickoski
Director, Nuclear Regulatory Affairs

Enclosures:

1. Evaluation of Proposed Change (Proprietary Version)
2. Evaluation of Proposed Change (Non-Proprietary Version)
3. Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to the Watts Bar Units 1 and 2 Nuclear Plants (Proprietary Version)
4. Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to the Watts Bar Units 1 and 2 Nuclear Plants (Non-Proprietary Version)
5. Westinghouse Electric Company LLC Application for Withholding Proprietary Information From Public Disclosure (Affidavit CAW-21-5138)

cc: See Page 4

cc (Enclosures):

NRC Regional Administrator - Region II
NRC 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

Enclosure 1

Evaluation of Proposed Change (Proprietary Version)

Subject: Application to Implement the FULL SPECTRUM LOCA (FSLOCA) Methodology for Loss-of-Coolant Accident (LOCA) Analysis," and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04)

Enclosure 2

Evaluation of Proposed Change (Non-Proprietary Version)

Subject: Application to Implement the FULL SPECTRUM LOCA (FSLOCA) Methodology for Loss-of-Coolant Accident (LOCA) Analysis," and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04)

Enclosure 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNITS 1 AND 2

Evaluation of Proposed Change (Non-Proprietary Version)

Subject: Application to Implement the FULL SPECTRUM LOCA (FSLOCA) Methodology for Loss-of-Coolant Accident (LOCA) Analysis," and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04)

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7. Proposed License Condition (Mark-Ups) for WBN Unit 2
8. Proposed License Condition (Final Typed) for WBN Unit 2

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit," Tennessee Valley Authority (TVA) is requesting a license amendment to the Watts Bar Nuclear Plant (WBN) Units 1 and 2. The proposed change will revise WBN Units 1 and 2 Technical Specification (TS) 5.9.5, "Core Operating Limits Report," to replace the loss-of-coolant accident (LOCA) analysis evaluation model references with reference to the FULL SPECTRUM™[1] Loss-of-Coolant Accident (FSLOCA™[1]) Evaluation Model analysis applicable to WBN Units 1 and 2, with replacement steam generators. The proposed change would also revise WBN Unit 1 TS 4.2.1, "Fuel Assemblies," to delete discussion of Zircalloy fuel rods. The proposed change also revises the WBN Unit 2 Operating License (OL) condition 2.C(4) to reflect the implementation of the FSLOCA Evaluation Model methodology.

2.0 DETAILED DESCRIPTION

2.1 Need for Proposed Change

TVA is requesting approval to use the FSLOCA Evaluation Model to evaluate the peak cladding temperatures for large-break and small-break LOCAs (LBLOCA and SBLOCA) (Reference 1). The use of the FSLOCA Evaluation Model results in a reduction in the peak cladding temperature in analyses of LBLOCA and SBLOCA. TVA is also requesting approval to use separate simulations performed in accordance with the FSLOCA Evaluation Model as part of the new Tritium Producing Burnable Absorber Rod (TPBAR) stress analysis methodology developed by Pacific Northwest National Laboratory and Westinghouse to provide a recovery of margin in the post-LOCA criticality evaluation in the presence of assumed TPBAR failures. The TPBARs are conservatively assumed to rupture due to high cladding temperature and pressure differential during LBLOCA events in the current licensing basis in the WBN dual-unit Updated Final Safety Analysis Report (UFSAR) Chapter 15 Appendix 15B. TPBAR rupture results in a positive reactivity addition and is a penalty in the post-LOCA criticality evaluation.

Approval of this license amendment request (LAR) will authorize the use of the FSLOCA Evaluation Model to evaluate the peak cladding temperatures for LBLOCA and SBLOCA. TVA proposes to use the new LOCA-specific TPBAR stress analysis methodology to evaluate the integrity of the TPBARs for the conditions expected during a LBLOCA. The results show that TPBARs will not rupture (with high probability and confidence). The continued integrity of the TPBARs results will be utilized in the core reload design process to simplify core designs, increase tritium production, and improve fuel cycle economics. The SBLOCA fuel rod thermal response predicted by the Westinghouse FSLOCA Evaluation Model application to WBN Units 1 and 2 is

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characterized by lower cladding temperature and higher external pressure than the LBLOCA response. The TPBAR cladding stress intensities during the SBLOCA are bounded by the LBLOCA and require no additional evaluation. The safety analysis process for each reload design will continue to demonstrate that all regulatory criteria are met.

2.2 Proposed Change

The following subsections describe each proposed TS change and its basis.

TS 4.2.1, "Fuel Assemblies" (WBN Unit 1 only)

The proposed change revises WBN Unit 1 TS 4.2.1, "Fuel Assemblies," to delete discussion of Zircalloy fuel rods. The FSLOCA Evaluation Model analysis considered ZIRLO® cladding. Insertion of Zircalloy fuel rods would require additional analysis and calculations. This change is consistent with the NRC approval of the FSLOCA Evaluation Model methodology (Reference 1).

TS 5.9.5, “Core Operating Limits Report” (WBN Units 1 and 2)

The proposed change revises TS 5.9.5, “Core Operating Limits Report,” to replace the LOCA analysis evaluation model references with reference to FSLOCA Evaluation Model analysis applicable to WBN Units 1 and 2, with replacement steam generators (SGs). This change reflects the FSLOCA Evaluation Model analyses performed for WBN Units 1 and 2 (see Enclosure 3).

OL condition 2.C(4) (WBN Unit 2 only)

The proposed change revises WBN Unit 2 OL condition 2.C(4) to add the statement: “FULL SPECTRUM LOCA Methodology shall be implemented when the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.”

Attachment 1 to Enclosure 1 provides the existing WBN Unit 1 TS pages marked-up to show the proposed changes. Attachment 2 to Enclosure 1 provides the existing WBN Unit 2 TS pages marked-up to show the proposed changes. Attachment 3 to Enclosure 1 provides the retyped WBN Unit 1 TS pages incorporating the proposed changes. Attachment 4 to Enclosure 1 provides the retyped WBN Unit 2 TS pages incorporating the proposed changes. Attachment 5 to Enclosure 1 provides the existing WBN Unit 1 TS Bases pages marked-up to show the proposed changes. Attachment 6 to Enclosure 1 provides the existing WBN Unit 2 TS Bases pages marked-up to show the proposed changes. Changes to the existing TS Bases are provided for information only and will be implemented under the Technical Specification Bases Control Program. Attachment 7 to Enclosure 1 shows the proposed changes to WBN Unit 2 OL condition 2.C(4) associated with the implementation of the FSLOCA Evaluation Model methodology. Attachment 8 to Enclosure 1 provides the WBN Unit 2 OL condition 2.C(4) retyped to show the changes incorporated.

2.3 Condition Intended to Resolve

The proposed changes will allow TVA to use the FSLOCA Evaluation Model to evaluate the peak cladding temperatures for LBLOCAs and SBLOCAs for WBN Units 1 and 2.

3.0 BACKGROUND

The Department of Energy (DOE) and TVA have agreed to cooperate in a program to produce tritium for the National Security Stockpile by irradiating TPBARs at WBN Units 1 and 2.

TPBARs are similar to standard burnable poison rod assemblies (BPRAs) inserted into fuel assemblies. The BPRAs absorb excess neutrons, and help control the power in the reactor to ensure an even power distribution and extend the time between refueling outages. TPBARs function in a matter similar to a BPRA, but TPBARs absorb neutrons using lithium aluminate instead of boron. Tritium is produced when the neutrons strike the lithium material. A solid zirconium material in the TPBAR (called a "getter") captures the produced tritium. Most of the tritium is contained within the TPBAR. However, a small fraction of the tritium will permeate through the TPBAR cladding into the reactor coolant system. After the TPBARs are removed from the core and shipped to a DOE extraction facility, the TPBARs are heated in a vacuum at high temperature to extract the tritium.

NRC has authorized TVA to irradiate up to 1,792 TPBARs in WBN Units 1 and 2 with the issuance of WBN Unit 1 License Amendment 107 (Reference 2) and WBN Unit 2 License Amendment 27 (Reference 3).

As described in this LAR, TVA will use separate FSLOCA Evaluation Model results with the new LOCA-specific TPBAR stress analysis methodology to evaluate the integrity of the TPBARs for the conditions expected during a LBLOCA and provide a recovery of margin in the post-LOCA criticality evaluation in the presence of assumed TPBAR failures. The continued integrity of the TPBARs results will be utilized in the core reload design process to simplify core designs, increase tritium production, and improve fuel cycle economics. The safety analysis process for each reload design will continue to demonstrate that all regulatory criteria are met (Reference 4).

4.0 TECHNICAL EVALUATION

There are two proposed methodology changes associated with this LAR:

- A change to reference the NRC-approved Westinghouse FSLOCA Evaluation Model (Reference 1) in WBN Units 1 and 2 TS 5.9.5 and the application of that methodology to WBN Units 1 and 2 (see Enclosure 3), as the new analyses of record for the LBLOCA and SBLOCA design basis scenarios.
- A change to the post-LOCA criticality evaluation methodology to credit the negative reactivity of TPBARs based on a structural integrity analysis of the TPBARs following a LBLOCA.

The application of a new LOCA-specific TPBAR stress analysis methodology provides a recovery of margin in the post-LOCA criticality evaluation by demonstrating that TPBARs will not rupture following a LBLOCA and are thus present as a negative reactivity source. In the current licensing basis (i.e., UFSAR Chapter 15 Appendix 15B), the TPBARs are conservatively assumed to rupture due to high cladding temperature and pressure differential during LBLOCA events. Separate effects test data show that in the event of TPBAR rupture, the release (or loss) of lithium aluminate pellet material can occur and would create a positive reactivity insertion. As a result, an assumption on the amount of positive reactivity from TPBAR rupture is included in the post-LOCA criticality evaluation. With the new technical approach in this licensing submittal that relies on LBLOCA simulations consistent with the FSLOCA Evaluation Model methodology and applies a LOCA-specific TPBAR stress analysis methodology, it can be demonstrated that the TPBARs will not rupture (with high probability and confidence). The continued integrity of the TPBARs will be credited in the core reload design process to simplify core designs, increase tritium production, and improve fuel cycle economics. The safety analysis process for each reload design will continue to demonstrate that all regulatory criteria are met.

Section 4.1 of this enclosure provides background information on the current NRC-approved TPBAR licensing submittals and the purpose of this LAR. Except for the reference to and application of (1) the FSLOCA Evaluation Model for the design basis small- and LBLOCA analyses and (2) the new LOCA-specific TPBAR stress analysis methodology, and the resulting credit for post-LOCA TPBAR integrity, there are no other changes to the safety analyses for WBN Units 1 and 2. The LOCA-specific TPBAR stress analysis **[** **]**^{a,c} the TPBAR failure would cause an unacceptable increase in reactivity in a conservative analysis. Towards the end of the fuel cycle, it is possible that TPBAR failure can be accommodated in the post-LOCA criticality analysis as a result of core reactivity depletion, lithium depletion, and lower initial reactor coolant system boron concentration. In such a circumstance, the LOCA-specific stress analysis **[** **]** ^{a,c} when TPBAR failure would be unacceptable, and TPBAR failure would be assumed at later times, consistent with the existing methodology. Alternatively, the LOCA-specific TPBAR stress analysis **[** **]**^{a,c} demonstrating that the TPBAR failure will not occur following an LBLOCA, and thus the negative reactivity is creditable at all times. The latter approach is followed in the demonstration presented in Section 4.3.2.

Section 4.2 of this enclosure presents the methodologies that will be used by TVA.

Section 4.3 of this enclosure presents a demonstration of applying the LOCA-specific TPBAR stress analysis and post-LOCA criticality methodologies to representative reload core designs at WBN Units 1 and 2. Conservative results are shown for evaluation of TPBAR integrity. Demonstration results for the post-LOCA criticality evaluation are shown for a typical reload core design. During the reload core safety analysis, these results will be shown to be applicable, or a similar approach will be followed using the new methodologies to confirm that safety analysis limits are met.

Section 4.4 of this enclosure presents a summary and conclusions of the technical evaluation that supports the proposed change.

4.1 Background

Insertion of TPBARs into WBN Units 1 and 2 presents the potential for a positive reactivity insertion following a LOCA in the event of cladding rupture at high temperatures. During core uncovery following a LOCA, the overheating of fuel rods causes radiant heating of the TPBARs located in adjacent control rod guide tubes. The heating of the TPBARs can result in rupture of the TPBAR cladding due to the increase in internal pressure, the decrease in external pressure, and the decrease in cladding strength at high temperature. Other potential cladding failure modes exist (see Section 4.2.2), but those failure modes are not limiting. The consequences of TPBAR cladding rupture are the potential for immediate expulsion of pellet (Lithium-6 (Li-6)) material from the TPBAR internals in the vicinity of the rupture location, and the potential for subsequent leaching of Li-6 in the long-term. The potential for loss of Li-6 from TPBARs by these mechanisms is based on out-of-pile testing and is assumed to be possible during or following a LOCA, in the worst case. Li-6 is a neutron absorbing material, and so a loss of Li-6 results in a positive reactivity addition.

In the WBN Units 1 and 2 post-LOCA criticality analyses, TPBAR failure could not be ruled out because of the predicted high fuel rod temperatures. Therefore, a conservative assumption was made that all TPBARs fail following a LBLOCA, and a conservative loss of Li-6 was assumed in the post-LOCA criticality analysis. The resulting positive reactivity addition was included in the post-LOCA criticality evaluation performed by Westinghouse to support core reload designs. The post-LOCA criticality evaluation (see Section 4.2.3) addresses a change in the core boron concentration following certain LBLOCA locations for the short-term and long-term phases of the LOCA response. The evaluation also considers the contribution of the xenon inventory at the time of the LOCA and its time-dependent behavior. A post-LOCA criticality evaluation is required for all Westinghouse plants, and for WBN Units 1 and 2, the potential for TPBAR failure is included in the assessment.

The new LOCA-specific TPBAR stress analysis methodology (see Section 4.2.2) relies on conditions resulting from LBLOCA simulations generated according to the FSLOCA Evaluation Model in a new approach to evaluate TPBAR integrity following a LOCA. The application of the FSLOCA Evaluation Model that is used for evaluation of TPBAR integrity is separate from the FSLOCA Evaluation Model application that demonstrates compliance with 10 CFR 50.46.

The application of the new stress analysis methodology demonstrates that TPBAR integrity will be maintained following a LBLOCA. As a result, the presence of intact TPBARs is credited in the post-LOCA criticality evaluation as a negative reactivity contribution. Application of the new LOCA-specific TPBAR stress analysis methodology requires TPBAR survival **[**
][a,c] TPBAR failure is of no consequence and survival does not need to be demonstrated. This approach simplifies the overall evaluation of post-LOCA criticality.

The improved results in the post-LOCA criticality evaluation allow core reload designs to be designed with fewer feed assemblies, lower enrichment, and fewer burnable absorbers, which will increase tritium production, and improve fuel cycle economics. The conservative evaluation methodology confirms that an acceptable margin to post-LOCA criticality is maintained for the entire fuel cycle.

4.2 Methodology

The three methodologies are: Westinghouse FSLOCA Evaluation Model Methodology (Section 4.2.1 of this enclosure); LOCA-Specific TPBAR Stress Analysis Methodology (Section 4.2.2 of this enclosure); and Westinghouse Post-LOCA Criticality Methodology (Section 4.2.3 of this enclosure).

4.2.1 Westinghouse FSLOCA Evaluation Model Methodology

This LAR includes a change to WBN Units 1 and 2 TS 5.9.5 to add the NRC-approved FSLOCA Evaluation Model methodology. This methodology produces separate results for both the LBLOCA break spectrum and the SBLOCA break spectrum. Enclosure 3 is the summary of the application of the FSLOCA Evaluation Model to WBN Units 1 and 2 that will become the new WBN dual-unit UFSAR analysis of record.[2] The advanced FSLOCA Evaluation Model methodology provides a reduction in the peak cladding temperature compared to the current LOCA methodologies (see Enclosure 3).

4.2.2 LOCA-Specific TPBAR Stress Analysis Methodology

A LOCA-specific TPBAR cladding stress analysis methodology has been developed by Pacific Northwest National Laboratory and Westinghouse to evaluate the structural integrity of TPBARs during a LOCA event. The TPBAR pressure boundary is represented by the coated cladding tube and the end plug weld joint. The objective of the methodology is to determine the potential for TPBAR cladding mechanical rupture under LBLOCA temperature and differential pressure conditions.

Methodology Overview

The stress analysis of the TPBAR following a large break LOCA is performed assuming conditions as calculated using the WCOBRA/TRAC-TF2 (WCT-TF2) code, the thermal-hydraulic code associated with the FSLOCA Evaluation Model. The purpose of the WCT-TF2 code is to predict the response of fuel rods (cladding temperatures and oxidation) following a postulated LOCA. In the LOCA-specific TPBAR stress analysis methodology, the WCT-TF2 code is used as it is within the FSLOCA Evaluation Model, but the predicted cladding temperatures are used as a surrogate for the TPBAR cladding temperature response. The TPBAR cladding temperatures are then used for the purpose of determining whether the TPBAR would be expected to rupture following the postulated LOCA.

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2 For WBN Unit 2 the FSLOCA Evaluation Model will become the analysis of record after the steam generators are replaced.

The stress field on the TPBAR is evaluated using a methodology derived using guidance from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) (Reference 5), including calculations for the primary membrane stresses (i.e., internal pressure) and the primary bending stresses (i.e., lateral deformation). The analytical expressions for the different stress contributions are evaluated and the stress intensities are computed for comparison to allowable stress limits. Demonstration that the allowable stress limits are met using the new methodology ensures that TPBAR integrity is maintained during a LOCA.

Two secondary stress components were considered and determined to be non-limiting. First, the cladding stresses arising from thermal shock during reflood quench (the transient radial temperature gradient) were evaluated using ANSYS.[3] The ANSYS analysis used a cooling rate and internal pressure that bounds the reflood heat transfer and fuel rod thermal conditions expected during a LBLOCA. The result of the evaluation is that the TPBAR cladding burst stress is well above (by more than 25,000 psi) the combined reflood quench and internal pressure stresses. In addition, steam corrosion data in the literature shows that the stainless steel (SS) 316 alloy will retain some level of ductility when exposed to temperatures encountered during a LBLOCA. The presence of ductility following steam corrosion of TPBAR coated cladding outer surface precludes TPBAR failure during reflood quench. Based on these considerations, the thermal stresses arising from reflood quench conditions do not challenge TPBAR survivability during a LBLOCA, and are not discussed further in this LAR. Second, the stress arising from the end plug to cladding tube weld joint geometric discontinuity along with internal pressure was also assessed using ANSYS. The results of the ANSYS analysis show that the analytical model used in the TPBAR design process produces a highly conservative estimate of the stress concentration (by a factor of ~2) at the weld joint. The mechanical analysis results are consistent with the measured burst pressure for the end plug weld joint qualification tests where rupture occurred in the cladding tube region some distance removed from the end plug weld joint. This evaluation demonstrates that cladding tube burst rupture (primary membrane and bending stress intensity) in regions separate from a qualified end plug weld joint would occur prior to failure of a TPBAR end plug during a LBLOCA. Therefore, the secondary stress intensity associated with the end plug discontinuity stresses has been determined to be non-limiting, and is not discussed further in this LAR.

Following a LBLOCA, the heating of the fuel cladding during the blowdown phase is rapid and is primarily due to the transfer of fuel pellet stored energy to the cladding after heat transfer to the surrounding fluid is interrupted during the blowdown phase. This rapid heating mechanism is not applicable to the TPBAR cladding and consequently the TPBAR cladding temperature lags the fuel rod cladding temperature due to the low initial temperature, heat capacity, and the time required for the associated heat transfer processes to occur. Based on those considerations and supporting analyses, **[**

][a,c]

3 ANSYS is a finite element analysis software used to simulate engineering problems.

The SBLOCA fuel rod thermal response predicted by the Westinghouse FSLOCA Evaluation Model application to WBN Units 1 and 2 is characterized by lower cladding temperature and higher external pressure. The TPBAR cladding stress intensities during the SBLOCA are bounded by the LBLOCA. The previous demonstration of survival for SBLOCA has been confirmed within the context of FSLOCA results and is not discussed further in this LAR.

<u>TPBAR Cladding Stress Evaluation for LBLOCA</u>

In the methodology used to perform the TPBAR cladding stress analysis, analytical expressions for the different load contributions are used to calculate the individual stress components that contribute to potential TPBAR cladding burst rupture. The stress intensities are computed from these stress components with a methodology that was developed using guidance provided in the ASME BPVC, Section III, Subsection NG, Article 3200. The TPBAR cladding stress intensities are calculated as a function of temperature and internal gas pressure. The TPBAR cladding temperatures are conservatively assumed to be equal to the fuel rod with the maximum temperature in the most limiting host fuel assembly. The fuel rod temperatures of interest are the transient peak cladding temperature and the transient axial cladding temperature distribution. The transient peak cladding temperature is used to calculate the cladding burst stress (allowable stress limit). The transient fuel rod cladding axial temperature distribution is used to compute the TPBAR internal gas pressure using the axial distribution of the internal gas volume within the TPBAR.

The initial gas pressure in the TPBAR at the start of the LOCA is based on a calculation of the helium generation in the TPBAR **[ ]**^{a,c} Helium is a by-product of the absorption of neutrons by Li-6 and the creation of tritium. The tritium generated during the fuel cycle is used to compute the helium gas inventory and thereby the initial TPBAR internal gas pressure. The analysis assumes 100 percent of the helium created by neutron absorption is released to the TPBAR internal void volume, and a bounding tritium production is assumed **[ ]**^{a,c} A second contributor to the internal gas inventory is the transient release of tritium from the lithium aluminate pellets. A certain fraction of the tritium generated during normal operation remains within the pellet. The remainder of the tritium generated is captured in the nickel-plated zirconium getter and not available to be release in gaseous form. During the high temperature period of a LOCA, this tritium is assumed to be released from the $LiAlO_2$ pellets in the form of either diatomic tritium (T_2) gas moles or as tritiated water (T_2O) vapor moles. These additional gas moles will increase the internal gas pressure. The TPBAR cladding stress analysis assumes that a bounding value of the tritium produced **[ ]**^{a,c} will be released into the internal gas volume of the TPBAR and contribute to pressurization of the cladding during a LOCA.

The potential for TPBAR failure due to thermal creep rupture is also included in the TPBAR cladding failure evaluation. Thermal creep rupture is a time-dependent mechanism of material deformation that leads to ductility exhaustion. A creep damage model was developed from high temperature time-to-failure tests conducted on TPBAR coated cold-worked SS-316 cladding. These tests were performed over a temperature range between 1500°F and 1625°F with rupture times that varied from 50 to 2850 seconds. A modified Larson-Miller model (Reference 6) was used to evaluate the dependency between applied stresses, temperature, and rupture time. The thermal creep rupture calculation is performed for all elevations of the TPBAR based on the local temperature response.

TPBAR Cladding Acceptance Criteria

The acceptance criteria for the TPBAR cladding stress analysis includes two modes: rapid burst and thermal creep rupture. The first allowable stress limit focuses on the condition that over-pressurization will lead to a rapid deformation process, commonly called burst rupture. The second allowable stress limit is concerned with the process of thermal creep rupture, which is a time-dependent mechanism of material deformation that leads to ductility exhaustion. In reviewing the material properties data in the ASME BPVC for the TPBAR cold-worked SS-316 cladding material, information on the stress intensities or ultimate tensile strengths is not available for the high temperature range of interest during a LOCA event. As a result, allowable stress limits were developed from open literature mechanical properties tests and TPBAR coated cladding mechanical testing.

The acceptance criterion for the rapid burst failure mode was developed using data from unirradiated cold-worked SS-316 available in the open literature and from tests performed by PNNL on unirradiated TPBAR coated cladding. Using data derived from unirradiated material produces a lower bound estimate of the burst stress of irradiated material based on information available in the literature. Reported mechanical testing on material from the Fast Flux Test Facility program found that irradiation of cold-worked SS-316 similar to TPBAR cladding material leads to an increase in both the yield stress and the ultimate tensile strength at pressurized water reactor (PWR) operating temperature conditions. An increase in ultimate tensile strength has also been observed in other SS-316 alloys used in light water reactor (LWR) reactor components.

For a given temperature, burst rupture will occur as the differential pressure increases to the stress intensity allowable limit. Once the cladding stress intensity exceeds the allowable limit, the material will rapidly fail by plastic deformation. Figure 4.2.2-1 shows the high temperature TPBAR cladding burst test data from coated and uncoated SS-316 tubing that have been used to establish the burst criterion (i.e., stress) as a function of temperature. The database includes a variety of tests performed on both coated and uncoated cold-worked SS-316 material under closed-end tube burst conditions. A majority of the burst tests were performed at temperatures above 1200°F (649 °C) at outer surface heating rates between 10°F/second to 20°F/second. The temperature and heating rate conditions are representative of those expected during a postulated LBLOCA for WBN Units 1 and 2. The lower bound thick-wall hoop stress at burst (dashed line) is selected as the burst stress acceptance criterion. The best estimate fit polynomial curve is shown for comparison.

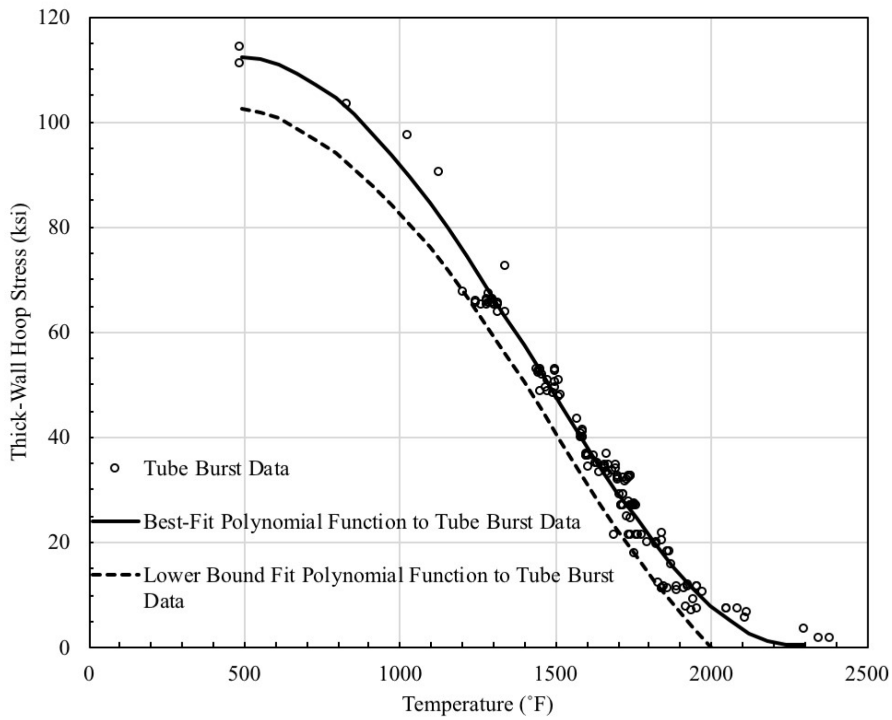

Figure 4.2.2-1: High Temperature Burst Stress Data and TPBAR Cladding Acceptance Criterion

The acceptance criterion for the thermal creep rupture failure mode was developed as follows. Curves of cladding hoop stress as a function of temperature for a constant time to rupture using a lower bound approach were constructed. A life fraction rule is applied to calculate the creep damage accumulation and a factor of safety for creep rupture.

For the rapid burst failure mode, the figure of merit is the primary membrane and bending stress safety factor, defined as the ratio of the stress at which rupture would occur and the stress intensity. For this figure of merit, a minimum value is limiting and a value of 1.0 or lower indicates failure (rupture).

For the thermal creep rupture mode, the figure of merit is a cumulative creep damage ratio, for which a maximum value is limiting and a value of 1.0 or higher indicates failure (rupture).

<u>Methodology Conservatisms</u>

Table 4.2.2-1 identifies the conservative elements and assumptions used in the LOCA-specific TPBAR stress analysis methodology that ensure a conservative evaluation of TPBAR structural integrity following a LOCA.

Table 4.2.2-1: TPBAR Structural Integrity Methodology Conservatisms

Description	Conservatism	Impact
Use fuel rod temperatures for TPBAR temperature and internal pressure	Fuel rod temperatures will be higher than a TPBAR by 100-200°F (Note 1)	~10 percent increase in stress intensity ~20-40 percent decrease in allowable stress
TPBAR internal void volume	Use minimum internal void volume for pressure calculation	Greater than 5 percent increase in stress intensity
Cladding tolerance stack up	Worst case fabrication tolerance values used in stress analysis	~2 percent increase in stress intensity
Cladding corrosion allowance	End of life corrosion thickness on both inner surface and outer surface.	~1 percent increase in stress intensity
Tritium released from the pellet available for gas pressurization	Assume 50 percent of tritium produced is in the pellet available for release	~5 percent increase in internal gas inventory
Burst criterion	Use lower bound of the burst stress data as allowable limit	~10-25 percent reduction in allowable stress

Note 1: Estimated maximum difference in temperature during the LOCA. The difference between the fuel rod and TPBAR cladding temperatures will decrease at times later in the event.

Application of the FSLOCA Evaluation Model to TPBAR Structural Integrity Analysis

As discussed in Section 4.1, a unique set of LBLOCA simulations (i.e., separate from the 10 CFR 50.46 compliance evaluation discussed in Enclosure 3) performed in accordance with the FSLOCA Evaluation Model is used as part of the LOCA-specific TPBAR stress analysis. The resulting conditions are then used to determine the response of the TPBAR cladding. A combined WBN Units 1 and 2 analysis is performed that is applicable to both units with replacement SGs.

A statistical approach similar to the approach used in FSLOCA Evaluation Model applications for the purpose of demonstrating compliance with the 10 CFR 50.46 acceptance criteria is used for the LOCA-specific TPBAR stress analysis. A Monte Carlo style uncertainty analysis is performed, and tolerance limits are constructed for the figures of merit related to the TPBAR structural integrity: 1) rupture due to primary membrane and bending stresses, and 2) rupture due to creep damage. The derived tolerance limits are compared to acceptance criteria (failure thresholds) to confirm compliance (structural integrity).

The figures-of-merit are the TPBAR primary membrane bending and stress safety factor, defined as the local burst stress divided by the local stress intensity, and the creep damage ratio (safety factor). The acceptance criteria that ensure TPBAR structural integrity are a primary membrane bending and stress safety factor greater than 1.0, and creep damage ratio less than 1.0. The results of applying the methodology support a conclusion that 95% of the population of postulated LBLOCAs will not result in TPBAR burst, with 95% confidence (consistent with the FSLOCA methodology). Consistent with the as-approved FSLOCA Evaluation Model, the TPBAR structural integrity analysis is performed for offsite power available (OPA) and loss-of-offsite power (LOOP) conditions.

The TPBAR cladding temperature is conservatively assumed to be equal to the fuel rod cladding temperature in an adjacent fuel rod, which represents the most limiting host assembly. The TPBAR cladding temperature is used to determine the degradation in cladding strength, using the established burst criterion, as discussed above. Assuming the TPBAR cladding temperature equals the adjacent fuel rod cladding temperature, the transient fuel rod cladding axial temperature distribution is used as a surrogate to calculate the TPBAR internal pressure throughout the transient, considering a weighted average of the temperature. The weighted average is computed by combining the axial temperature distribution and the axial distribution of the internal gas volume to yield a volume-weighted temperature. Helium is a byproduct of the tritium production reaction. The initial internal TPBAR pressure at the start of the LOCA is based on a calculation of the helium production in the TPBAR as a function of a bounding tritium production
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4.2.3 Westinghouse Post-LOCA Criticality Methodology

Post-LOCA conditions in the reactor core must be evaluated to address the potential for the following four scenarios to challenge the margin to criticality:

1) Following the LOCA blowdown some volume of water at the initial primary system boron concentration will remain in the reactor vessel. The initial boron concentration changes during the fuel cycle and can be a dilution source compared to emergency core cooling system (ECCS) water sources. During the reflooding phase these water sources must be evaluated to address the potential for criticality.

2) Following a LBLOCA in the cold leg, the reactor vessel will evolve to a boiling pot mode of core cooling. The steam exiting the top of the core exits the break, condenses, and accumulates in the containment sump. Over time this process decreases the sump boron concentration. Then, as the sump water becomes the ECCS injection source the delivery of that water into the reactor vessel must be evaluated to address the potential for criticality. This must be evaluated at the time

of alignment of cold leg recirculation when the refueling water storage tank is depleted.

3) The time for alignment of hot leg recirculation, currently three hours for WBN Units 1 and 2, is determined by analysis to prevent unacceptable results. Alignment for hot leg recirculation is the mitigation action that prevents the occurrence of post-LOCA criticality. Subsequent to alignment for hot leg recirculation, the boron concentration in the sump will approach the boron concentration in the reactor vessel, and the dilution effect will be resolved.

4) In the long-term, the negative reactivity associated with the xenon present in the reactor core will decay to zero. The loss of the negative xenon reactivity must be shown to not challenge criticality.

The current WBN Units 1 and 2 licensing basis (i.e., UFSAR Chapter 15 Appendix 15B) has justified that only two of the above scenarios challenge post-LOCA criticality. These are the evaluation of the hot leg alignment scenario and the long-term xenon-free scenario; however, all scenarios are confirmed as part of each cycle's standard reload calculations. The methodology used in the current WBN Units 1 and 2, licensing basis remains unchanged except for the proposed licensing basis (with supporting technical justification) that the TPBARs do not fail during a LBLOCA. The following conservative assumptions and inputs are elements of the post-LOCA criticality methodology:

- Minimum refueling water storage tank boron concentration
- Minimum cold leg accumulator boron concentration
- Minimum ice condenser boron concentration
- Minimum reactor coolant system boron concentration consistent with peak xenon
- Minimum containment sump boron concentration vs. time
- Time of hot leg recirculation alignment = 3 hours
- Conservative xenon reactivity for hot leg recirculation case
- Zero xenon reactivity for long-term case
- Cold conditions (50°F to 212°F)
- Most reactive time in life
- No credit for control rod insertion negative reactivity
- No credit for void feedback negative reactivity

The new post-LOCA criticality methodology assumes no TPBAR failure during the time-in-cycle that the failure contributes to the criticality evaluation.

4.3 Results

Results are presented for the FSLOCA Evaluation Model for 10 CFR 50.46 compliance and the structural integrity evaluation. These results represent a demonstration of applying the three methodologies to support future reload core designs at WBN Units 1 and 2. Results are shown for the application of these methodologies. During the reload core safety analysis, these results will be shown to be applicable or a revised analysis will be performed using the new methodologies, as appropriate.

4.3.1 FSLOCA Evaluation Model Results for 10 CFR 50.46 Compliance

The FSLOCA Evaluation Model analysis presented in Enclosure 3 demonstrates that there is a high level of probability that the following criteria in 10 CFR 50.46 are met:

(b)(1) The analysis peak clad temperature (PCT) corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Because the resulting PCT is less than 2,200°F, the analysis with the FSLOCA Evaluation Model confirms that 10 CFR 50.46 acceptance criterion (b)(1) (i.e., “Peak Cladding Temperature less than 2,200°F”) is demonstrated. The results are shown in Table 4 in Enclosure 3.

(b)(2) The analysis maximum local oxidation (MLO) corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Because the resulting MLO is less than 17 percent when converting the time-at-temperature to an equivalent cladding reacted using the Baker-Just correlation and adding the pre-transient corrosion, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2) (i.e., “Maximum Local Oxidation of the cladding less than 17 percent”) is demonstrated. The results are shown in Table 4 in Enclosure 3.

(b)(3) The analysis core-wide oxidation (CWO) corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. Because the resulting CWO is less than 1 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3) (i.e., “Core-Wide Oxidation less than 1 percent”) is demonstrated. The results are shown in Table 4 in Enclosure 3.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains in a coolable geometry. This criterion is met by demonstrating compliance with criteria (b)(1), (b)(2), and (b)(3), and by assuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed. Criteria (b)(1), (b)(2), and (b)(3) have been met for WBN Units 1 and 2, as shown in Table 4 in Enclosure 3.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of the continued delivery of cooling water to the core. The actions that are currently in place to maintain long-term cooling are not impacted by the application of the NRC-approved FSLOCA Evaluation Model (Reference 1).

Based on the analysis results for Region I and Region II presented in Table 4 in Enclosure 3, it is concluded that WBN Units 1 and 2, comply with the criteria in 10 CFR 50.46.

Comparison to Results from Analyses of Record

The existing SBLOCA and LBLOCA analysis-of-record (AOR) results for Watts Bar Units 1 and 2 are presented in UFSAR Tables 15.3-2 and 15.4-18.

The SBLOCA AOR results originate from analyses performed with an evaluation model developed according to Appendix K of 10 CFR Part 50. The FSLOCA EM is a best-estimate plus uncertainty method, which relaxes some of the conservatisms required for EMs developed to Appendix K of 10 CFR Part 50. The improvement in the SBLOCA analysis PCT results is primarily attributed to the more realistic treatment of various phenomena within the FSLOCA EM, most notably the decay heat modeling; the prior analyses assumed decay heat based on 1.2 times ANSI/ANS 5.1-1971, whereas the analysis with the FSLOCA EM is based on ANSI/ANS 5.1-1979.

The SBLOCA MLO results are substantially higher for the analysis with the FSLOCA EM than in the AORs. This is primarily attributed to the AOR results only considering the oxidation accrued during the LOCA transient, whereas the results from the FSLOCA EM include the steady-state corrosion. The CWO results for both the AORs and the analysis with the FSLOCA EM indicate little-to-no core-wide oxidation during the SBLOCA transient.

The improvement in the LBLOCA PCT results are attributed to differences between the FSLOCA EM and the legacy evaluation models, including, but not limited to, the following: improvements to the statistical analysis method (elimination of the superposition penalty (Unit 1 only) and larger sample size (both units)); improvements to the fuel temperature calibration; improvements to the axial power shape methodology; and improvements to the swelling, burst, and blockage models.

The LBLOCA MLO results for the Watt Bar Unit 1 AOR are higher compared to the results from the analysis with the FSLOCA EM, while the AOR results for Watt Bar Unit 2 are lower. The AOR results only consider the oxidation accrued during the LOCA transient, whereas the results from the FSLOCA EM include the steady-state corrosion. The increase in the MLO for Watt Bar Unit 2 is primarily attributed to the contribution of the steady-state corrosion. The Watt Bar Unit 1 AOR MLO result is based on a LOCA transient with a cladding temperature in excess of the PCT result and for which the time-at-temperature has been artificially increased (both artifacts of the AOR evaluation model). As such, the excessive transient cladding temperature and artificial increase in time-at-temperature from the AOR evaluation model leads to a conservatively high result.

The LBLOCA CWO results for the Watt Bar Unit 2 AOR and the analysis with the FSLOCA EM are relatively low. For the Watt Bar Unit 1 AOR, the CWO result is based on the same transient as the local oxidation result with a conservatively high cladding temperature. As such, the reduction in CWO for the analysis with the FSLOCA EM in comparison with the Watt Bar Unit 1 AOR is largely attributed to the use of a transient with lower temperature, as predicted by differences in the evaluation models.

4.3.2 TPBAR Structural Integrity Analysis Results

Enclosure 3 describes an analysis performed for WBN Units 1 and 2, using the FSLOCA Evaluation Model to satisfy the 10 CFR 50.46 acceptance criteria. For the separate TPBAR structural integrity analysis, the same WCT-TF2 input model was used. Also, the same major plant parameter and analysis assumptions (i.e., Tables 1 through 3 in Enclosure 3) were used. However, in the TPBAR structural integrity analysis, it was assumed that **[**

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The results in Table 4.3.2-1 show that the 95/95 TPBAR structural integrity analysis results retain significant margin to the acceptance criteria for the OPA analysis (safety factor = 1.94), and for the LOOP analysis (safety factor = 1.96). Creep damage is negligible under both conditions. A detailed discussion of the results for the LOOP analysis follows.

Figure 4.3.2-1 shows the fuel rod peak cladding temperature resulting from the WCT-TF2 simulations assuming LOOP as a function of effective break area (geometrical break area plus the effect of uncertainty applied to the critical flow model). The results indicate behavior typical of simulations performed in accordance with the FSLOCA Evaluation Model.

Figure 4.3.2-2 shows the minimum primary membrane and bending stress safety factor results from all of the simulated LOCAs in the LOOP analysis **[**

]^{a,c} as shown in Table 4.3.2-2. Increased tritium mass is a dominant contributor to more limiting results, as it produces a higher TPBAR internal pressure, and thus more limiting stress conditions. For this analysis as a simplification, TPBAR survival is assumed to be required/credited **[**

]^{a,c} Note that application of the proposed methodology requires TPBAR survival **[**

]^{a,c}**,** TPBAR failure is of no consequence and survival does not need to be demonstrated as long as post-LOCA criticality margins remain acceptable in the presence of ruptured TPBARs.

Details of the LOCA scenario resulting in the 95/95 lower tolerance limit for primary membrane and bending stress safety factor are shown in Figures 4.3.2-3 through 4.3.2-5. Figure 4.3.2-3 shows the temperature response of the TPBAR. The TPBAR peak cladding temperature corresponds with the fuel rod peak cladding temperature, as it is assumed that the TPBAR cladding temperature equals the fuel rod cladding temperature. The TPBAR volume-weighted average temperature reflects the axial temperature distribution of the TPBAR throughout the transient as it is calculated for the purpose of determining the TPBAR internal pressure. At approximately 30 seconds after the break, the average temperature decreases with time as the refill and reflood of the core region quench the lower portions of the rod and the hot assembly liquid level recovers (see Figure 4.3.2-4).

Figure 4.3.2-5 shows the calculated burst stress (the stress required to rupture the TPBAR at the present temperature conditions), the stress intensity, and the ratio of the two as the calculated safety factor at the same axial node, at the node yielding the lowest safety factor. As described in Section 4.2.2, the fuel rod cladding temperature is not a realistic representation of the TPBAR temperature during the blowdown phase,
[

][a,c] The calculated stress intensity reflects two main effects; first, the general behavior of the TPBAR peak cladding temperature is evident by the similarities with Figure 4.3.2-3, where increases in peak temperature result in increases in stress intensity. Similarly, the effect of the reduction in TPBAR average temperature, a surrogate for internal pressure and starting around 30 seconds after the break, results in a general reduction in stress intensity over the same period. Conversely, the burst stress follows the opposite trend with respect to TPBAR peak cladding temperature, where increased local temperatures lead directly to a reduction in stress required to cause rupture. The resulting safety factor reaches a minimum around 35 seconds after the break, when TPBAR peak temperatures remain high and the lower portions of the core have not yet been recovered.

In summary, the 95/95 tolerance limits for (1) the primary membrane and bending stress safety factor, and (2) the creep damage ratio maintain significant margin to the burst stress failure criteria, and as such there is very high confidence that the TPBARs will not rupture following a postulated LBLOCA accident.

The WCT-TF2 code was exercised in a manner consistent with the NRC-approved FSLOCA Evaluation Model (Reference 1). The NRC's Safety Evaluation Report contains 15 limitations and conditions. The limitations and conditions were reviewed for applicability to the TPBAR structural integrity analysis, and no compliance issues were identified.

The treatment for the uncertainty in the gamma energy redistribution is discussed on pages 29-75 and 29-76 WCAP-16996-P-A, Revision 1 (Reference 1), and the equation for the assumed increase in hot rod and hot assembly relative power is presented on page 29-76. The power increase in the hot rod and hot assembly due to energy redistribution in the FSLOCA EM simulations supporting the TPBAR structural integrity analysis was calculated incorrectly. This error resulted in a 0% to 5% deficiency in the modeled hot rod and hot assembly rod linear heat rates on a run-specific basis, depending on the as-sampled value for the uncertainty. The effect of the error correction was evaluated against the application of the FSLOCA EM to the Watts Bar Units 1 and 2 TPBAR structural integrity analysis.

The error correction has only a limited impact on the power modeled for a single assembly in the core. As such, there is a negligible impact of the error correction on the system thermal-hydraulic response during the postulated LOCA.

Parametric PWR sensitivity studies, derived from a subset of uncertainty analysis simulations covering various design features and fuel arrays, were examined to determine the sensitivity of the analysis results to the error correction. The magnitude of the impact from the error correction was found to be different for the different transient phases (i.e., blowdown versus reflood) based on the PWR sensitivity studies and existing power distribution sensitivity studies. Based on the results from the PWR sensitivity studies, the correction of the error is estimated to result in a fuel cladding temperature increase of 31°F for the time period relevant to TPBAR structural integrity, which is assumed to also lead to a TPBAR cladding temperature increase of 31°F.

To estimate the effect of changes in the TPBAR cladding temperature on the results of the TPBAR structural integrity analysis, the TPBAR structural integrity calculations were performed with an assumed increase in TPBAR temperature throughout the transient. The updated TPBAR structural integrity analysis results, including the error correction, are determined using a TPBAR cladding temperature increase of 31°F and are presented in Table 4.3.2-1.

Table 4.3.2-1: TPBAR Structural Integrity Analysis Results

Outcome	Criterion	Offsite Power Available (OPA)	Loss of Offsite Power (LOOP)
95/95 Primary membrane and bending stress safety factor	> 1.0	1.94*	1.96*
95/95 Cumulative creep damage ratio	< 1.0	0.0	0.0
* The results presented in this table reflect the correction of the error in gamma energy redistribution uncertainty. The initial results for primary membrane and bending stress safety factor were 2.04 and 2.07 for OPA and LOOP, respectively, prior to the correction. The cumulative creep damage ratio is negligibly affected by the correction. The figures presenting the analysis results correspond to the initial results.			

Table 4.3.2-2: Maximum Tritium Production per TPBAR

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]a,c

Figure 4.3.2-1: Peak Fuel Rod Cladding Temperature in WCT-TF2 Simulations Assuming LOOP

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]a,c

Figure 4.3.2-2: Primary Membrane and Bending Stress Safety Factor Results [
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]a,c

Figure 4.3.2–3: TPBAR Peak Cladding Temperature and Volume-Weighted Average Temperature for the OPA Analysis Resulting in the 95/95 Minimum Safety Factor

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]a,c

Figure 4.3.2-4: Hot Assembly Collapsed Liquid Level and Volume-Weighted Average Temperature for the OPA Analysis Resulting in the 95/95 Minimum Safety Factor

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]a,c

Figure 4.3.2-5: Stress Intensity, Burst Stress, and Safety Factor Results for the OPA Analysis Resulting in the 95/95 Minimum Safety Factor

4.3.3 Post-LOCA Criticality Results

The demonstration of TPBAR structural integrity, as discussed in Section 4.3.2, support the changes to the Post-LOCA Criticality Methodology described in Section 4.2.3. The new post-LOCA criticality methodology assumes no TPBAR failure during the time-in-cycle that the failure contributes to the criticality evaluation. Positive reactivity additions associated with TBPAR failure have been eliminated. As a consequence, post-LOCA subcriticality margin is increased. The standard reload methodology will confirm post-LOCA subcriticality is maintained each cycle.

4.4 Summary and Conclusions

This section presents a summary and conclusions of the technical evaluation that supports the proposed change.

FSLOCA Evaluation Model Application for 10 CFR 50.46 Compliance

The Region I SBLOCA analysis for WBN Units 1 and 2, was performed in accordance with the NRC-approved FSLOCA Evaluation Model methodology. The limiting transient is a cold leg break with a break diameter of 4.2 inches. The limiting peak cladding temperature result is 976°F. The limiting maximum local oxidation result is 8.9 percent.[4] The maximum core wide oxidation result is zero percent. The regulatory acceptance criteria of 10 CFR 50.46 were met.

The Region II LBLOCA analysis for WBN Units 1 and 2, was performed in accordance with the NRC-approved FSLOCA Evaluation Model methodology. The limiting peak cladding temperature result is 1457°F. The limiting maximum local oxidation result is 8.9 percent.[4] The maximum core wide oxidation result is zero percent. The regulatory acceptance criteria of 10 CFR 50.46 were met.

TPBAR Structural Integrity

A new LOCA-specific TPBAR structural integrity analysis methodology has been developed. The TPBAR cladding failure modes that are applicable to the LOCA scenarios have been evaluated. The results of the analysis presented in Section 4.3 show that TPBARs will not rupture (with high probability and confidence) following an LBLOCA.

Post-LOCA Criticality

The current licensing basis (References 2 and 3) post-LOCA criticality evaluations have conservatively assumed that 100 percent of the TPBARs will experience failure due to cladding overheating and burst. Cladding failure would result in a positive reactivity addition due to loss of Li-6 from the TPBAR internals. The results of the TPBAR structural integrity analysis have demonstrated that no TPBAR failures will occur

[4] Due to the low amounts of predicted transient oxidation, the 95/95 MLO results are comprised of pre-transient oxidation only.

following a design basis LOCA event (with high probability and confidence). As a consequence, post-LOCA subcriticality margin is increased. The standard reload methodology will confirm post-LOCA subcriticality is maintained.

5.0 REGULATORY EVALUATION

5.1 Applicable Regulatory Requirements and Criteria

The LOCA evaluations are described in the following sections of the WBN dual-unit UFSAR:

- Section 15.1.8.4, “Distribution of Decay Heat Following Loss of Coolant Accident”
- Table 15.1-2, “Summary of Initial Conditions and Computer Codes Used”
- Section 15.3.1, “Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuate the Emergency Core Cooling System”
- Section 15.4.1, “Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)”
- Appendix 15B, “Operation with a Tritium Production Core”

For the LOCA analyses, the principal reviews performed by NRC for WBN Unit 1 are documented in the safety evaluation for WBN Unit 1 License Amendment 21 (Reference 7). The principal reviews performed by NRC for WBN Unit 2 are documented in the Safety Evaluation Report (SER), NUREG-0847 Supplement 24 (Reference 8).

For the TPBARs assessments, the principal reviews performed by NRC for WBN Unit 1 are documented in the safety evaluation for WBN Unit 1 License Amendment 107 (Reference 2). The principal reviews performed by NRC for WBN Unit 2 are documented in the safety evaluation for WBN Unit 2 License Amendment 27 (Reference 3).

General Design Criteria

WBN Units 1 and 2 were designed to meet the intent of the "Proposed General Design Criteria for Nuclear Power Plant Construction Permits" published in July, 1967. The Watts Bar construction permit was issued in January 1973. The UFSAR, however, addresses the NRC General Design Criteria (GDC) published as Appendix A to 10 CFR 50 in July 1971, including Criterion 4 as amended October 27, 1987.

Each criterion listed below is followed by a discussion of the design features and procedures that meet the intent of the criteria. Any exception to the 1971 GDC resulting from the earlier commitments is identified in the discussion of the corresponding criterion.

GDC 35 - Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Compliance with GDC 35 is described in Section 3.1.2.4 of the WBN UFSAR. The ECCS is described in Section 6.3 of the WBN UFSAR.

NRC Generic Letter (GL) 88-16

GL 88-16, "Removal of Cycle-Specific, Parameter Limits from Technical Specifications," dated October 4, 1988, provides that it is acceptable for licensees to control reactor physics parameter limits by specifying an NRC-approved calculation methodology. These parameter limits may be removed from the TS and placed in a cycle-specific Core Operating Limits Report (COLR) that is required to be submitted to the NRC every operating cycle or each time it is revised. Consistent with the guidance in NRC GL 88-16, WBN Units 1 and 2 TS 5.9.5, "Core Operating Limits Report (COLR)" requires the following:

- The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC.
- The COLR, including any midcycle revisions or supplements, is to be provided upon issuance for each reload cycle to the NRC.
- The COLR is defined in Section 1.1 of the TS.
- The TS include a list of references of the NRC-approved methodologies that are used to determine the cycle-specific core operating limits. TS 5.9.5 identifies the NRC-approved analytical methodologies that are used to determine the core operating limits for WBN Units 1 and 2.

Upon approval of the proposed LAR, the guidance in the GL 88-16 continues to be met because the proposed change will continue to specify the NRC-approved methodologies used to determine the core operating limits.

5.2 Precedents

This LAR is similar to one approved by the NRC for the Diablo Canyon Nuclear Power Plant (ML19312C440 and ML19316A109), which revised Diablo Canyon TS 5.6.5b , "Core Operating Limits Report (COLR), " to replace the existing LOCA methodologies with the NRC-approved LOCA methodology contained in WCAP-16996-P-A, Revision 1.

5.3 Significant Hazard Consideration

Tennessee Valley Authority (TVA) proposes to revise the current licensing basis of Facility Operating License Nos. NPF-90 and NPF-96 for Watts Bar Nuclear Plant (WBN) Units 1 and 2. The proposed change would revise WBN Units 1 and 2 Technical Specification (TS) 5.9.5, "Core Operating Limits Report," to replace the loss-of-coolant accident (LOCA) analysis evaluation model references with reference to the FULL SPECTRUM™ Loss-of-Coolant Accident (FSLOCA™)[5] Evaluation Model analysis applicable to both WBN Unit 1 and Unit 2 with replacement steam generators. The proposed change would also revise WBN Unit 1 TS 4.2.1, "Fuel Assemblies," to delete discussion of Zircalloy fuel rods. The proposed change also revises the WBN Unit 2 Operating License (OL) condition 2.C(4) to reflect the implementation of the FSLOCA Evaluation Model methodology.

TVA is requesting approval to use the FSLOCA Evaluation Model to evaluate the peak cladding temperatures for LBLOCA and SBLOCA . The use of the FSLOCA Evaluation Model results in a reduction in the peak cladding temperature in analyses of LBLOCA and SBLOCA. TVA is also requesting approval to use the FSLOCA Evaluation Model results (i.e., reduction in peak cladding temperature) with the new Pacific Northwest National Laboratory Tritium Producing Burnable Absorber Rod (TPBAR) stress analysis methodology to provide a recovery of margin in the post-LOCA criticality evaluation in the presence of assumed TPBAR failures. The TPBARs were conservatively assumed to rupture due to high cladding temperature and pressure differential during LBLOCA events. TPBAR rupture results in a positive reactivity addition and is a penalty in the post-LOCA criticality evaluation. TVA proposes to use the FSLOCA Evaluation Model methodology and the new LOCA-specific TPBAR stress analysis methodology to evaluate the integrity of the TPBARs. The results show that TPBARs will not rupture (with high probability and confidence). The continued integrity of the TPBARs results will be utilized in the core reload design process to simplify core designs, increase tritium production, and improve fuel cycle economics. The safety analysis process for each reload design will continue to demonstrate that all regulatory criteria are met.

This proposed change will support the TPBAR irradiation plans for WBN Units 1 and 2 to support national security needs. TVA has concluded that the changes to WBN Unit 1 TSs 4.2.1 and 5.9.5 and the WBN Unit 2 TS 5.9.5 do not involve a significant hazards consideration. TVA's conclusion is based on its evaluation in accordance with 10 CFR 50.91(a)(1) of the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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5 FULL SPECTRUM and FSLOCA are trademarks of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

1. *Does the proposed amendment involve a significant increase in the probability or consequence of an accident previously evaluated?*

   Response: No.

   The proposed change to WBN Units 1 and 2 TS 5.9.5, "Core Operating Limits Report," to replace the LOCA analysis evaluation model references with reference to the FSLOCA Evaluation Model analysis applicable to both WBN Unit 1 and Unit 2 with replacement steam generators. The proposed change would also revise WBN Unit 1 TS 4.2.1, "Fuel Assemblies," to delete discussion of Zircalloy fuel rods. These changes implement a Nuclear Regulatory Commission (NRC) approved LOCA evaluation model. The analysis results for WBN Units 1 and 2, based on using the new evaluation model meet the regulatory requirements of 10 CFR 50.46. The use of a new NRC-approved LOCA evaluation model will not increase the potential for an accident. Therefore, the possibility of an accident is not increased by the proposed changes. Because the reactor core meets the regulatory requirements of 10 CFR 50.46 after a postulated LOCA, the consequences of an accident are not increased by the proposed changes.

   The use of separate simulations performed in accordance with the FSLOCA Evaluation Model as part of the new TPBAR stress analysis methodology developed by Pacific Northwest National Laboratory and Westinghouse provide a recovery of margin in the post-LOCA criticality evaluation in the presence of assumed TPBR failures. The TPBARs were conservatively assumed to rupture due to high cladding temperature and pressure differential during LBLOCA events. TPBAR rupture results in a positive reactivity addition and is a penalty in the post-LOCA criticality evaluation. TVA proposes to use the FSLOCA Evaluation Model methodology and the new LOCA-specific TPBAR stress analysis methodology to evaluate the integrity of the TPBARs. The results show that TPBARs will not rupture (with high probability and confidence). Crediting the continued integrity of the TPBARs results will be utilized in the core reload design process to simplify core designs, increase tritium production, and improve fuel cycle economics. The safety analysis process for each reload design will continue to demonstrate that all regulatory criteria are met. The use of a new TPBAR stress analysis methodology will not increase the potential for an accident. Therefore, the possibility of an accident is not increased by the proposed changes. Because the TPBAR failure analysis results show that TPBARs will not rupture (with high probability and confidence), the consequences of an accident are not increased by the proposed changes.

   Based on the above discussions, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

2. *Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?*

   Response: No.

   The proposed change to WBN Units 1 and 2 TS 5.9.5 to replace the LOCA analysis evaluation model references with reference to FSLOCA Evaluation Model and the corresponding change to WBN Unit 1 TS 4.2.1 implement an NRC-approved LOCA evaluation model. The use of the new TPBAR stress analysis methodology to analyze the potential for TPBAR failures provides recovery of margin in the post-LOCA criticality evaluation in the presence of assumed TPBAR failures. The use of these two new analytical methodologies will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does the proposed amendment involve a significant reduction in a margin of safety?*

   Response: No.

   The proposed change to WBN Units 1 and 2 TS 5.9.5 to replace the LOCA analysis evaluation model references with reference to the FSLOCA Evaluation Model and the corresponding change to WBN Unit 1 TS 4.2.1 implement an NRC-approved LOCA evaluation model. The analysis results for WBN Units 1 and 2, based on using the new evaluation model, meet the regulatory requirements of 10 CFR 50.46 with increased margin after a postulated LOCA.

   The analysis results for WBN Units 1 and 2, based on using the new LOCA specific TPBAR stress analysis methodology show that TPBARs will not rupture (with high probability and confidence), which provides an increase of margin in the post-LOCA criticality evaluation in the presence of assumed TPBAR failures.

   Accordingly, the proposed changes do not involve a significant reduction in a margin of safety.

5.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the license amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. WCAP-16996-P-A, Revision 1, “Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA METHODOLOGY),” dated November 2016 (ML17277A130)
2. 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)
3. NRC Letter to TVA, “Watts Bar Nuclear Plant, Units 1 and 2- Issuance of Amendment Regarding Revision to Watts Bar Nuclear Plant, Unit 2, Technical Specification 4.2.1, ‘Fuel Assemblies,’ and Watts Bar Nuclear Plant, Units 1 and 2, Technical Specifications Related To Fuel Storage (EPID L-2017-LLA-0427),” dated May 22, 2019 (ML18347B330)
4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985.
5. 2010 ASME Boiler Pressure Vessel Code Section III Division 1 Subsection NG
6. F.R. Larson and J. Miller, “A Time-Temperature Relationship for Rupture and Creep Stresses,” Transactions of the ASME, Vol 74, July 1952, p 765–775.
7. NRC Letter to TVA, “Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Best Estimate Large Break Loss-of-Coolant Accident Analysis, TS-98-016 (TAC No. MA6038),” dated March 17, 2000 (ML003693761)
8. NUREG-0847, Supplement 24, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant Unit 2,” September 2011 (ML11277A148)

ATTACHMENT 1

Proposed TS Changes (Mark-Ups) for WBN Unit 1

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries

The site and exclusion area boundaries shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2 (within the 3-mile circle).

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of ~~Zircalloy,~~ ZIRLO®~~,~~ or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. For Unit 1, Watts Bar is authorized to place a maximum of 1792 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

4.2.2 Control Rod Assemblies

The reactor core shall contain 57 control rod assemblies. The control material shall be either silver-indium-cadmium or boron carbide with silver indium cadmium tips as approved by the NRC.

(continued)

5.9 Reporting Requirements (continued)

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 Limit
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 percent of rated thermal power is specified in a previously approved method, 100.6 percent of rated thermal power 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 6 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102 percent of rated thermal power (3411 MWt) shall be used.

The approved analytical methods are specifically those described in the following documents:

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

2~~a~~. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~WCAP-12945-P-A, Volume I (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate Loss of Coolant Analysis," March 1998 (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).~~

~~b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985. Addendum 2, Rev. 1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997. (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear~~

(continued)

~~Enthalpy Rise Hot Channel Factor).~~

(continued)

ATTACHMENT 2

Proposed TS Changes (Mark-Ups) for WBN Unit 2

5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration).

2~~a~~. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005 (W Proprietary).(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).~~

~~2b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985. Addendum 2, Rev. 1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997. (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).~~

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

(continued)

ATTACHMENT 3

Proposed TS Changes (Final Typed) for WBN Unit 1

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries

The site and exclusion area boundaries shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2 (within the 3-mile circle).

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of ZIRLO® or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. For Unit 1, Watts Bar is authorized to place a maximum of 1792 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

4.2.2 Control Rod Assemblies

The reactor core shall contain 57 control rod assemblies. The control material shall be either silver-indium-cadmium or boron carbide with silver indium cadmium tips as approved by the NRC.

(continued)

5.9 Reporting Requirements (continued)

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 Limit
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 percent of rated thermal power is specified in a previously approved method, 100.6 percent of rated thermal power 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 6 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102 percent of rated thermal power (3411 MWt) shall be used.

The approved analytical methods are specifically those described in the following documents:

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

(continued)

ATTACHMENT 4

Proposed TS Changes (Final Typed) for WBN Unit 2

5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration).

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

(continued)

ATTACHMENT 5

Proposed TS Bases Changes (Mark-Ups) for WBN Unit 1

(for information only)

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident (LOCA), the 10 CFR 50.46 acceptance criteria must be met~~peak cladding temperature must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks~~ (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. ~~Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.~~

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a loss of coolant accident (LOCA), the 10 CFR 50.46 acceptance criteria must be met~~peak cladding temperature (PCT) must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks~~ (Ref. 3);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. Refer to the Bases for the Reactor Core Safety Limits, B 2.1.1 for a discussion of the applicable DNBR limits. The W-3 Correlation with a DNBR limit of 1.3, or the ABB-NV correlation with a DNBR limit of 1.13, is applied in the heated region below the first mixing vane grid. In addition, the W-3 or WLOP DNB correlations are applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation for VANTAGE 5H and VANTAGE+ fuel or the WRB-2M correlation for RFA-2 fuel with IFMs. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3. For system pressures in the range of 185 to 1800 psia, the WLOP correlation DNBR limit is 1.18.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Application of these criteria provides assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analyses that verify compliance with the 10 CFR 50.46 acceptance criteria~~the acceptability of the resulting peak cladding temperature~~ (Ref. 3) model $F^N_{\Delta H}$ as well as the Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using either the Movable Incore Detector System or the PDMS (Ref. 5). Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Policy Statement.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a ~~large break~~ loss of coolant accident, the 10 CFR 50.46 criteria must be met~~peak cladding temperature must not exceed 2200°F~~ (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), rod group alignment, sequence, overlap, and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and compliance with the 10 CFR 50.46 acceptance criteria (Ref. 21)~~for limiting peak clad temperature to < 2200°F)~~; and

2. Boration to ensure recovery and maintenance of SDM (k_{eff} < 1.0).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation;
- Containment Vent Isolation;
- Reactor Trip;
- Turbine Trip;
- Feedwater Isolation;
- Start of all auxiliary feedwater (AFW) pumps;
- Control room ventilation isolation; and
- Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) suction to containment sump.

These other functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the turbine and reactor to limit power generation;
- Isolation of main feedwater (MFW) to limit secondary side mass losses

;

(continued)

BASES

REFERENCES (continued)

5. Code of Federal Regulations, Title 10, Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

6. WCAP-12096, Rev. 7, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," March 1997.

7. WCAP-10271-P-A, Supplement 1 and Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System." May 1986 and June 1990.

8. Watts Bar Technical Requirements Manual, Section 3.3.2, "Engineered Safety Feature Response Times."

9. TVA Letter to NRC, November 9, 1984, "Request for Exemption of Quarterly Slave Relay Testing, (L44 841109 808)."

10. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar, Westinghouse letter to TVA WAT-D-10128.

11. Westinghouse letter to TVA (WAT-D-8347), September 25, 1990, "Charging/Letdown Isolation Transients" (T33 911231 810).

12. Design Change Notice W-38238 associated documentation.

13. WCAP-13877, Rev. 1, “Reliability Assessment of Westinghouse Type AR Relays Used As SSPS Slave Relays,” August 1998.

14. TVA’s Letter to NRC dated February 25, 2000, “WBN Unit 1 Request for TS Amendment for TS 3.3.2 - ESFAS Instrumentation.”

15. WCAP-13632-P-A Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements,” January 1996.

16. WCAP-14036-P-A, Revision 1, “Elimination of Periodic Protection Channel Response Time Tests,” October 1998.

17. WCAP-14333-P-A, Revision 1, “Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,” October 1998.

18. WCAP-15376-P-A, Revision 1, “Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,” March 2003.

19. Westinghouse letter to TVA, WAT-D-11248, “Revised Justification for Applicability of Instrumentation Technical Specification Improvements to the Automatic Switchover to Containment Sump Signal,” June 2004.

20. Letter from John G. Lamb (NRC) to Mr. Preston D. Swafford (TVA) dated March 4, 2009, Includes Enclosures (a) Amendment No. 75 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1 and (b) NRC Safety Evaluation (SE) for Amendment No. 75.

21. Code of Federal Regulations, Title 10, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

(continued)

BASES

BACKGROUND (continued)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. Although not required for accident mitigation, the valves will automatically open as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is also considered to determine if it yields limiting results. The loss of offsite power assumption imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large break LOCA is a ~~double ended guillotine~~ break in the cold leg. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting (for loss of offsite power assumption) and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The ~~worst case~~ small break LOCA analyse~~e~~is also assumes a time delay before pumped flow is assumed to inject into the reactor coolant system~~reaches the core~~. For intermediate~~the larger range of small~~ breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. At very small break sizes, the safety injection pumps are capable of mitigating the inventory loss during the small-break LOCA, and the accumulators do not play a significant role in the accident mitigation.~~As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase~~.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46, Paragraph b (Ref. 3) will be met with a high level of probability following a LOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

~~For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. Both large and small-break analyses use a nominal accumulator line volume from the accumulator to the check valve. The safety analysis assumes accumulator water volumes~~values~~ of 7518 gallons and 8191 gallons. To allow for instrument inaccuracy, values of 7630 gallons and 8000 gallons are specified.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

~~The small break LOCA analysis is performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.~~ The maximum nitrogen cover pressure ~~analysis~~ limit ~~of 690 psig~~ prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. The LOCA analyses support a range of 585 to 690 psig. To account for the accumulator tank design pressure rating, and to allow for instrument accuracy values of ≥ 610 psig and ≤ 660 psig are specified for the pressure indicator in the main control room.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

(continued)

BACKGROUND (continued)

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence for a loss of offsite power. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46, Paragraph b (Ref. 2), will be met with a high level of probability following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The large break LOCA is the limiting case since the safety analysis assumes least negative reactivity insertion.

The upper limit on boron concentration of 3300 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 60°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The acceptable temperature range of 60°F to 105°F is assumed in the large and small-break LOCA analyses per approved methods (Ref. 3)~~break LOCA analysis, and the small break analysis value bounds the upper temperature limit of 105°F~~. The upper temperature limit of 105°F is also used in the containment OPERABILITY analysis. Exceeding the upper temperature limit could~~will~~ result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water following a LOCA~~ and higher containment pressures due to reduced containment spray cooling capacity~~. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of the NRC Policy Statement.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section 6.3, “Emergency Core Cooling System,” and Section 15.0, “Accident Analysis.”

2. Watts Bar Drawing 1-47W605-243, “Electrical Tech Spec Compliance Tables.”

3. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The containment was also designed for an external pressure load equivalent to 2.0 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was -0.1 psig. This resulted in a minimum pressure inside containment of 1.4 psig, which is less than the design load. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with WCAP-16996-P-A, Revision 1 ~~10 CFR 50, Appendix K~~ (Ref. 2). Containment pressure satisfies Criterion 2 of the NRC Policy Statement.
LCO	Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System or Air Return Fans.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODES 5 or 6.

(continued)

BASES (continued)

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour. When opening or closing Penetration 1-EQH-271-0010 or 1-EQH-271-0011 in the Shield Building Dome, the differential pressure between the Containment and the Annulus may exceed the equal to or greater than -0.1 and equal to or less than +0.3 psid requirement. During this operation, time is allowed for Containment/Annulus pressure equalization to be re-established.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits (≥ -0.1 and ≤ +0.3 psid relative to the annulus, value does not account for instrument error, Ref. 3) ensures that plant operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. Watts Bar FSAR, Section 6.2.1, "Containment Functional Design."

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."~~

3. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

APPLICABLE SAFETY ANALYSES (continued)

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure signal setpoint to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The Containment Spray System total response time of 221 seconds is composed of signal delay, diesel generator startup, and system startup time.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with WCAP-16996-P-A, Revision 1 ~~10 CFR 50, Appendix K~~ (Ref. 3).

Inadvertent actuation of the Containment Spray System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated steady state pressure differential relative to the Shield Building annulus is 1.4 psid, which is below the containment design external pressure load of 2.0 psid.

The Containment Spray System satisfies Criterion 3 of the NRC Policy Statement.

LCO

During a DBA, one train of Containment Spray System and RHR Spray System is required to provide the heat removal capability assumed in the safety analyses. To ensure that these requirements are met, two containment spray trains and two RHR spray trains must be OPERABLE with power from two safety related, independent power supplies. Therefore, in the event of an accident, at least one train in each system operates.

Each containment spray train typically includes a spray pump, header, valves, a heat exchanger, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and transferring suction to the containment sump. This suction path realignment is accomplished by manual operator action upon receipt of a Low-Low level alarm for the RWST.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criterion (GDC) 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal System," GDC 40, "Testing of Containment Heat Removal Systems, and GDC 50, "Containment Design Basis."

2. Watts Bar FSAR, Section 6.2, "Containment Systems."

3. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."~~

4. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

BASES

APPLICABLE SAFETY ANALYSES (continued)

of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure.For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with WCAP-16996-P-A, Revision 1~~10 CFR 50, Appendix K~~ (Ref. 2).

The modeled ARS actuation from the containment analysis is based upon a response time associated with exceeding the containment pressure High-High signal setpoint to achieving full ARS air flow. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The ARS total response time of 540 $\pm$ 60 seconds consists of the built in signal delay.

The ARS satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one train of the ARS is required to provide the minimum air recirculation for heat removal and hydrogen mixing assumed in the safety analyses. To ensure this requirement is met, two trains of the ARS must be OPERABLE. This will ensure that at least one train will operate, assuming the worst case single failure occurs, which is in the ESF power supply.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1 (continued)

fan and/or motor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency was developed considering the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.10.2

Verifying ARS fan motor current with the return air backdraft dampers closed confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of 92 days conforms with the testing requirements for similar ESF equipment and considers the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.10.3

Verifying the OPERABILITY of the air return damper to the proper opening torque (Ref. 3) provides assurance that the proper flow path will exist when the fan is started. By applying the correct torque to the damper shaft, the damper operation can be confirmed. The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.

REFERENCES

1. Watts Bar FSAR, Section 6.8, "Air Return Fans."

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."~~

3. System Description N3-30RB-4002.

APPLICABLE SAFETY ANALYSES (continued)

For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with WCAP-16996-P-A, Revision 1~~10 CFR 50, Appendix K~~ (Ref. 2). The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The ice bed satisfies Criterion 3 of the NRC Policy Statement.

LCO

The ice bed LCO requires the existence of the required quantity of stored ice, appropriate distribution of the ice and the ice bed, open flow paths through the ice bed, and appropriate chemical content and pH of the stored ice. The stored ice functions to absorb heat during a DBA, thereby limiting containment air temperature and pressure. The chemical content and pH of the ice provide core SDM (boron content) and remove radioactive iodine from the containment atmosphere when the melted ice is recirculated through the ECCS and the Containment Spray System, respectively.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice bed. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ice bed is not required to be OPERABLE in these MODES.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.6

This SR ensures that a representative sampling of ice baskets, which are relatively thin walled, perforated cylinders, have not been degraded by wear, cracks, corrosion, or other damage. Each ice basket must be raised at least 10 feet for this inspection. However, for baskets where vertical lifting height is restricted due to overhead obstruction, a camera shall be used to perform the inspection. The Frequency of 40 months for a visual inspection of the structural soundness of the ice baskets is based on engineering judgment and considers such factors as the thickness of the basket walls relative to corrosion rates expected in their service environment and the results of the long term ice storage testing.

SR 3.6.11.7

This SR ensures that initial ice fill and any subsequent ice additions meet the boron concentration and pH requirements of SR 3.6.11.5. The SR is modified by a NOTE that allows the chemical analysis to be performed on either the liquid or resulting ice of each sodium tetraborate solution prepared. If ice is obtained from offsite sources, then chemical analysis data must be obtained for the ice supplied.

REFERENCES

1. Watts Bar FSAR, Section 6.2, "Containment Systems"

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016~~Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models"~~

3. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

4. Westinghouse Letter, WAT-D-10686, “Upper Limit Ice Boron Concentration In Safety Analysis”

APPLICABLE SAFETY ANALYSES (continued)

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed with respect to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System and the ARS being rendered inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with WCAP-16996-P-A, Revision 1 ~~10 CFR 50, Appendix K~~ (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

An additional design requirement was imposed on the ice condenser door design for a small break accident in which the flow of heated air and steam is not sufficient to fully open the doors.

For this situation, the doors are designed so that all of the doors would partially open by approximately the same amount. Thus, the partially opened doors would modulate the flow so that each ice bay would receive an approximately equal fraction of the total flow.

This design feature ensures that the heated air and steam will not flow preferentially to some ice bays and deplete the ice there without utilizing the ice in the other bays.

In addition to calculating the overall peak containment pressures, the DBA analyses include the calculation of the transient differential pressures that would occur across subcompartment walls during the initial blowdown phase of the

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.12.7

Verifying, by visual inspection, that the top deck doors are in place, not obstructed, and verifying free movement of the vent assembly provides assurance that the doors are performing their function of keeping warm air out of the ice condenser during normal operation, and would not be obstructed if called upon to open in response to a DBA. The Frequency of 92 days is based on engineering judgment, which considered such factors as the following:

a. The relative inaccessibility and lack of traffic in the vicinity of the doors make it unlikely that a door would be inadvertently left open;

b. Excessive air leakage would be detected by temperature monitoring in the ice condenser; and

c. The light construction of the doors would ensure that, in the event of a DBA, air and gases passing through the ice condenser would find a flow path, even if a door were obstructed.

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis."

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."~~

3. TVA Letter to NRC dated July 31, 1996 - Proposed License Amendment - Containment Systems.

ATTACHMENT 6

Proposed TS Bases Changes (Mark-Ups) for WBN Unit 2

(for information only)

BASES (continued)

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident (LOCA), the 10 CFR 50.46 acceptance criteria must be met~~peak cladding temperature must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks~~ (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. ~~Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.~~

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a loss of coolant accident (LOCA), the 10 CFR 50.46 acceptance criteria must be met~~peak cladding temperature (PCT) must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks~~ (Ref. 3);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. Refer to the Bases for the Reactor Core Safety Limits, B 2.1.1, for a discussion of the applicable DNBR limits. The W-3 Correlation with a DNBR limit of 1.3 is applied in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is applied in the analysis of accident conditions where the system pressure is below the range of the WRB-2M correlation for RFA-2 fuel with IFMs. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3.

Application of these criteria provides assurance that the hottest fuel rod in the core does not experience a DNB.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analyses that verify compliance with the 10 CFR 50.46 acceptance criteria~~the acceptability of the resulting peak cladding temperature~~ (Ref. 3) model $F^N_{\Delta H}$ as well as the Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the PDMS (Ref. 4). Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the 10 CFR 50.46 criteria must be met~~peak cladding temperature must not exceed 2200°F~~ (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), rod group alignment, sequence, overlap, and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents.

ESFAS protection functions are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and compliance with the 10 CFR 50.46 acceptance criteria (Ref. 23)~~for limiting peak clad temperature to < 2200°F)~~; and

2. Boration to ensure recovery and maintenance of SDM (k_{eff} < 1.0).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation;
- Containment Vent Isolation;
- Reactor Trip;
- Turbine Trip;
- Feedwater Isolation;
- Start of all auxiliary feedwater (AFW) pumps;
- Control room ventilation isolation; and
- Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) suction to containment sump.

(continued)

BASES

REFERENCES (continued)

10. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar, Westinghouse letter to TVA WAT-D-10128.

11. Westinghouse letter to TVA (WAT-D-8347), September 25, 1990, "Charging/Letdown Isolation Transients" (T33 911231 810).

12. Unit 1 Design Change Notice W-38238 and Unit 2 Engineering Document Construction Release 53352 and associated documentation.

13. WCAP-13877-P-A, Revision 2, “Reliability Assessment of Westinghouse Type AR Relays Used As SSPS Slave Relays.”

14. Not Applicable for Unit 2

15. WCAP-13632-P-A Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements,” January 1996.

16. WCAP-14036-P-A, Revision 1, “Elimination of Periodic Protection Channel Response Time Tests,” October 1998.

17. WCAP-14333-P-A, Revision 1, “Probablistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,” October 1998

18. WCAP-15376-P-A, Revision 1, “Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,” March 2003

19. Westinghouse letter to TVA, WAT-D-11248, “Revised Justification for Applicability of Instrumentation Technical Specification Improvements to the Automatic Switchover to Containment Sump Signal,” June 2004.

20. Letter from John G. Lamb (NRC) to Mr. Preston D. Swafford (TVA) dated March 4, 2009, Includes Enclosures (a) Amendment No. 75 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1 and (b) NRC Safety Evaluation (SE) for Amendment No. 75.

21. Deleted

22. WCAP-13878-P-A, Revision 2, “Reliability Assessment of Potter & Brumfield MDR Series Relays.”

23. Code of Federal Regulations, Title 10, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

BACKGROUND (continued)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. Although not required for accident mitigation, the valves will automatically open as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is also considered to determine if it yields limiting results. The loss of offsite power assumption imposes a delay wherein the ECCS pumps cannot deliver flow until the diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large break LOCA is a ~~double ended guillotine~~ break in the cold leg. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting (for loss of offsite power assumption) and the pumps being loaded and delivering full flow. The delay time is conservatively set to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown phase of a large break LOCA.

The ~~worst case~~ small break LOCA analysi~~e~~s also assumes a time delay before pumped flow is assumed to inject into the reactor coolant system.~~reaches the core.~~ For ~~the larger range of small~~ intermediate breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. At very small break sizes, the safety injection pumps are capable of mitigating the inventory loss during the small-break LOCA, and the accumulators do not play a significant role in the accident mitigation.~~As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.~~

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46, Paragraph b (Ref. 3) will be met with a high level of probability following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

~~For the small break LOCA analysis, a nominal contained accumulator water volume of 7855 gallons is used, while a range of 7518 - 8191 gallons was used for the large break LOCA analysis.~~ The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. Both large and

(continued)

small-break analyses use a nominal accumulator line volume from the accumulator to the check valve. The safety analysis assumes accumulator water volumes of 7518 gallons and 8191 gallons. To allow for instrument inaccuracy, values of 7630 gallons and 8000 gallons are specified.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

~~The small break LOCA analysis is performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.~~ The maximum nitrogen cover pressure analysis limit of 690 psig prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. The LOCA analyses support a range of 585 psig to 690 psig. To account for the accumulator tank design pressure rating, and to allow for instrument accuracy values of ≥ 610 psig and ≤ 660 psig are specified for the pressure indicator in the main control room.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

(continued)

BASES

BACKGROUND (continued)

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence for a loss of offsite power. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the diesel generators (DGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46, Paragraph b (Ref. 2), will be met with a high level of probability following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 60°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The acceptable temperature range of 60°F to 105°F is assumed in the large break and small-break LOCA analyses per approved methods (Ref. 2).~~LOCA analysis, and the small break analysis value bounds the upper temperature limit of 105°F.~~ The upper temperature limit of 105°F is also used in the containment OPERABILITY analysis. Exceeding the upper temperature limit could~~will~~ result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water following a LOCA~~ and higher containment pressures due to reduced containment spray cooling capacity~~. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, “RCS Loops - MODE 5, Loops Filled,” and LCO 3.4.8, “RCS Loops - MODE 5, Loops Not Filled.” MODE 6 core cooling requirements are addressed by LCO 3.9.5, “Residual Heat Removal (RHR) and Coolant Circulation - High Water Level,” and LCO 3.9.6, “Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level.”

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. The specified temperature range is ≥ 60 °F and ≤ 105 °F and does not account for instrument error. The 24 hour Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

SR 3.5.4.2

The required minimum RWST water level is ≥ 370,000 gallons (value does not account for instrument error). Verification every 7 days of the presence of this water volume ensures that a sufficient initial supply of water is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. Watts Bar FSAR, Section 6.3, “Emergency Core Cooling System,” and Section 15.0, “Accident Analysis.”

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

(continued)

APPLICABLE SAFETY ANALYSES (continued)	The containment was also designed for an external pressure load equivalent to 2.0 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was -0.1 psig. This resulted in a minimum pressure inside containment of 1.4 psig, which is less than the design load. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with WCAP-16996-P-A, Revision 1 ~~10 CFR 50, Appendix K~~ (Ref. 2).. Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System or Air Return Fans.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODES 5 or 6.

(continued)

BASES

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, “Containment,” which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits (≥ -0.1 and ≤ +0.3 psid relative to the annulus, value does not account for instrument error) ensures that plant operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. Watts Bar FSAR, Section 6.2.2, “Containment Heat Removal Systems.”

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~Title 10, Code of Federal Regulations, Part 50, Appendix K, “ECCS Evaluation Models.”~~

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the ARS being rendered inoperable (Ref. 2).

The DBA analyses show that the maximum peak containment pressure of 9.36 psig results from the LOCA analysis, and is calculated to be less than the containment maximum allowable pressure of 15 psig. The maximum peak containment atmosphere temperature results from the SLB analysis. The calculated transient containment atmosphere temperatures are acceptable for the DBA SLB.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure signal setpoint to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The Containment Spray System total response time of 234 seconds is composed of signal delay, diesel generator startup, and system startup time.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with WCAP-16996-P-A, Revision 1~~10 CFR 50, Appendix K~~ (Ref. 3).

Inadvertent actuation of the Containment Spray System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated steady state pressure differential relative to the Shield Building annulus is 1.4 psid, which is below the containment design external pressure load of 2.0 psid.

The Containment Spray System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, “General Design Criterion (GDC) 38, “Containment Heat Removal,” GDC 39, “Inspection of Containment Heat Removal System,” GDC 40, “Testing of Containment Heat Removal Systems, and GDC 50, “Containment Design Basis.”

2. NPG-SDD-WBN2-72-4001, “Containment Heat Removal Spray System.”

3. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~Title 10, Code of Federal Regulations, Part 50, Appendix K, “ECCS Evaluation Models.”~~

4. American Society of Mechanical Engineers (ASME) OM Code, “Code for Operation and Maintenance of Nuclear Power Plants.”

BASES

BACKGROUND (continued)

purging all potential hydrogen pockets in containment. When the containment pressure falls below a predetermined value, the ARS fans are manually de-energized. Thereafter, the fans are manually cycled on and off if necessary to control any additional containment pressure transients.

The ARS also functions, after all the ice has melted, to circulate any steam still entering the lower compartment to the upper compartment where the Containment Spray System can cool it.

The ARS is an ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. The operation of the ARS, in conjunction with the ice bed, the Containment Spray System, and the Residual Heat Removal (RHR) System spray, provides the required heat removal capability to limit post accident conditions to less than the containment design values.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System, RHR System, and ARS being inoperable (Ref. 1). The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with WCAP-16996-P-A, Revision 1~~10 CFR 50, Appendix K~~ (Ref. 2).

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section 6.8, “Air Return Fans.”

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~Title 10, Code of Federal Regulations, Part 50, Appendix K, “ECCS Evaluation Models.”~~

3. System Description N3-30RB-4002.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with WCAP-16996-P-A, Revision 1~~10 CFR 50, Appendix K~~ (Ref. 2). The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature." In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs. The ice bed satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The ice bed LCO requires the existence of the required quantity of stored ice, appropriate distribution of the ice and the ice bed, open flow paths through the ice bed, and appropriate chemical content and pH of the stored ice. The stored ice functions to absorb heat during a DBA, thereby limiting containment air temperature and pressure. The chemical content and pH of the ice provide core SDM (boron content) and remove radioactive iodine from the containment atmosphere when the melted ice is recirculated through the ECCS and the Containment Spray System, respectively.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice bed. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ice bed is not required to be OPERABLE in these MODES.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.7

This SR ensures that initial ice fill and any subsequent ice additions meet the boron concentration and pH requirements of SR 3.6.11.5. The SR is modified by a NOTE that allows the chemical analysis to be performed on either the liquid or resulting ice of each sodium tetraborate solution prepared. If ice is obtained from offsite sources, then chemical analysis data must be obtained for the ice supplied.

REFERENCES

1. Watts Bar FSAR, Section 6.2, “Containment Systems” and Section 6.7, “Ice Condenser System.”

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~Title 10, Code of Federal Regulations, Part 50, Appendix K, “ECCS Evaluation Models.”~~

3. Westinghouse Letter, WAT-D-10686, “Upper Limit Ice Boron Concentration In Safety Analysis.”

APPLICABLE SAFETY ANALYSES (continued)	Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed with respect to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System and the ARS being rendered inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with WCAP-16996-P-A, Revision 1~~10 CFR 50, Appendix K~~ (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

An additional design requirement was imposed on the ice condenser door design for a small break accident in which the flow of heated air and steam is not sufficient to fully open the doors.

For this situation, the doors are designed so that all of the doors would partially open by approximately the same amount. Thus, the partially opened doors would modulate the flow so that each ice bay would receive an approximately equal fraction of the total flow.

This design feature ensures that the heated air and steam will not flow preferentially to some ice bays and deplete the ice there without utilizing the ice in the other bays.

In addition to calculating the overall peak containment pressures, the DBA analyses include the calculation of the transient differential pressures that would occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand the local transient pressure differentials for the limiting DBAs.

The ice condenser doors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.12.7

Verifying, by visual inspection, that the top deck doors are in place, not obstructed, and verifying free movement of the vent assembly provides assurance that the doors are performing their function of keeping warm air out of the ice condenser during normal operation, and would not be obstructed if called upon to open in response to a DBA. The Frequency of 92 days is based on engineering judgment, which considered such factors as the following:

a. The relative inaccessibility and lack of traffic in the vicinity of the doors make it unlikely that a door would be inadvertently left open;

b. Excessive air leakage would be detected by temperature monitoring in the ice condenser; and

c. The light construction of the doors would ensure that, in the event of a DBA, air and gases passing through the ice condenser would find a flow path, even if a door were obstructed.

REFERENCES

1. Watts Bar FSAR, Section 6.2.1, “Containment Functional Design” and Section 6.7, “Ice Condenser System.”

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.~~Title 10, Code of Federal Regulations, Part 50, Appendix K, “ECCS Evaluation Models.”~~

(continued)

ATTACHMENT 7

Proposed License Condition (Mark-Ups) for WBN Unit 2

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and to the rules regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

TVA is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 34, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) TVA shall implement permanent modifications to prevent overtopping of the embankments of the Fort Loudon Dam due to the Probable Maximum Flood by June 30, 2018.

(4) PAD4TCD may be used to establish core operating limits until the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1. FULL SPECTRUM LOCA Methodology shall be implemented when the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

(5) By December 31, 2019, the licensee shall report to the NRC that the actions to resolve the issues identified in Bulletin 2012-01, "Design Vulnerability in Electrical Power System," have been implemented.

(6) The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90 and 50.54(p).

(7) TVA shall fully implement and maintain in effect all provisions of the Commission approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The TVA approved CSP was discussed in NUREG-0847, Supplement 28, as amended by changes approved by License Amendment No. 7.

(8) TVA shall implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Report for the facility, as described in NUREG-0847, Supplement 29, subject to the following provision:

ATTACHMENT 8

Proposed License Condition (Final Typed) for WBN Unit 2

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and to the rules regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

TVA is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 34, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) TVA shall implement permanent modifications to prevent overtopping of the embankments of the Fort Loudon Dam due to the Probable Maximum Flood by June 30, 2018.

(4) PAD4TCD may be used to establish core operating limits until the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1. FULL SPECTRUM LOCA Methodology shall be implemented when the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

(5) By December 31, 2019, the licensee shall report to the NRC that the actions to resolve the issues identified in Bulletin 2012-01, "Design Vulnerability in Electrical Power System," have been implemented.

(6) The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90 and 50.54(p).

(7) TVA shall fully implement and maintain in effect all provisions of the Commission approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The TVA approved CSP was discussed in NUREG-0847, Supplement 28, as amended by changes approved by License Amendment No. 7.

(8) TVA shall implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Report for the facility, as described in NUREG-0847, Supplement 29, subject to the following provision:

Enclosure 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNITS 1 AND 2

Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to the Watts Bar Units 1 and 2 Nuclear Plants (Proprietary Version)

This document contains information that Westinghouse Electric Company LLC considers to be proprietary in nature.

Subject: Application to Implement the FULL SPECTRUM LOCA (FSLOCA) Methodology for Loss-of-Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04)

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNITS 1 AND 2

Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to the Watts Bar Units 1 and 2 Nuclear Plants (Non-Proprietary Version)

This document does not contain proprietary information.

Subject: Application to Implement the FULL SPECTRUM LOCA (FSLOCA) Methodology for Loss-of-Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04)

Attachment 1

Suggested Technical Evaluation Section of the Watts Bar Units 1 and 2 LAR Input

(50 pages, including cover page)

APPLICATION OF WESTINGHOUSE FULL SPECTRUM LOCA EVALUATION MODEL TO THE WATTS BAR UNITS 1 AND 2 NUCLEAR PLANTS

1.0 INTRODUCTION

An analysis with the **FULL SPECTRUM™** loss-of-coolant accident (**FSLOCA™**) evaluation model (EM) has been completed for the Watts Bar Units 1 and 2 Nuclear Plants. The FSLOCA EM (Reference 1) was developed to address the full spectrum of loss-of-coolant accidents (LOCAs) which result from a postulated break in the reactor coolant system (RCS) of a pressurized water reactor (PWR). The break sizes covered by the Westinghouse FSLOCA EM include any break size in which break flow is beyond the capacity of the normal charging pumps, up to and including a double ended guillotine (DEG) rupture of an RCS cold leg with a break flow area equal to two times the pipe area, including what traditionally are defined as Small and Large Break LOCAs.

The break size spectrum is divided into two regions. Region I includes breaks that are typically defined as Small Break LOCAs (SBLOCAs). Region II includes break sizes that are typically defined as Large Break LOCAs (LBLOCAs).

The FSLOCA EM explicitly considers the effects of fuel pellet thermal conductivity degradation (TCD) and other burnup-related effects by calibrating to fuel rod performance data input generated by the PAD5 code (Reference 2), which explicitly models TCD and is benchmarked to high burnup data in Reference 2. The fuel pellet thermal conductivity model in the WCOBRA/TRAC-TF2 code used in the FSLOCA EM explicitly accounts for pellet TCD.

Three of the Title 10 of the Code of Federal Regulations (CFR) 50.46 criteria (peak cladding temperature (PCT), maximum local oxidation (MLO), and core-wide oxidation (CWO)) are considered directly in the FSLOCA EM. A high probability statement is developed for the PCT, MLO, and CWO that is needed to demonstrate compliance with 10 CFR 50.46 acceptance criteria (b)(1), (b)(2), and (b)(3) (Reference 3) via statistical methods. The MLO is defined as the sum of pre-transient corrosion and transient oxidation consistent with the position in Information Notice 98-29 (Reference 4). The coolable geometry acceptance criterion, 10 CFR 50.46 (b)(4), is assured by compliance with acceptance criteria (b)(1), (b)(2), and (b)(3), and by demonstrating that fuel assembly grid deformation due to combined seismic and LOCA loads does not extend to the inboard fuel assemblies such that a coolable geometry is maintained.

The FSLOCA EM has been generically approved by the Nuclear Regulatory Commission (NRC) for Westinghouse 3-loop and 4-loop plants with cold leg Emergency Core Cooling System (ECCS) injection (Reference 1). Since Watts Bar Units 1 and 2 are Westinghouse designed 4-loop plants with cold leg ECCS injection, the approved method is applicable. Information required to address Limitations and Conditions 9 and 10 of the NRC's Safety Evaluation Report (SER) for Reference 1 was docketed in Reference 13 in support of application of the FSLOCA EM to Westinghouse 4-loop plants.

This report summarizes the application of the Westinghouse FSLOCA EM to Watts Bar Units 1 and 2. The application of the FSLOCA EM to Watts Bar Units 1 and 2 is consistent with the NRC-approved methodology (Reference 1), with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The application of the FSLOCA EM to Watts Bar Units 1 and 2 is consistent with the conditions and limitations as identified in the NRC's SER for Reference 1, and is also applicable for the

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Watts Bar Units 1 and 2 plant design and operating conditions. A single composite model was developed for both Watts Bar Units 1 and 2, assuming replacement steam generators (RSGs). The plant designs and operating parameters were assessed to create the composite model for a conservative application of the FSLOCA EM.

Both Tennessee Valley Authority and the analysis vendor (Westinghouse) have interface processes which identify plant configuration changes potentially impacting safety analyses. These interface processes, along with Westinghouse internal processes for assessing EM changes and errors, are used to identify the need for LOCA analysis impact assessments.

The major plant parameter and analysis assumptions used in the Watts Bar Units 1 and 2 analysis with the FSLOCA EM are provided in Tables 1 through 3b.

2.0 METHOD OF ANALYSIS

2.1 FULL SPECTRUM LOCA Evaluation Model Development

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 (Reference 3 and Reference 6) and Appendix K, "ECCS Evaluation Models," to permit the use of a realistic EM to analyze the performance of the ECCS during a hypothetical LOCA. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. After the rule change, Westinghouse developed and received approval for a best-estimate LBLOCA EM, which is discussed in Reference 8. This EM is referred to as the Code Qualification Document (CQD), and was developed following Regulatory Guide (RG) 1.157 (Reference 7). The CQD was the EM implemented for Watts Bar Unit 1. Westinghouse's approved best-estimate LBLOCA EM subsequent to the CQD EM is discussed in Reference 9. This EM is referred to as the Automated Statistical Treatment of Uncertainty Method (ASTRUM), also developed following RG 1.157 (Reference 7). ASTRUM was implemented as the EM for Watts Bar Unit 2.

When the FSLOCA EM was being developed, the NRC issued RG 1.203 (Reference 10) which expands on the principles of RG 1.157, while providing a more systematic approach to the development and assessment process of a PWR accident and safety analysis EM. Therefore, the development of the FSLOCA EM followed the Evaluation Model Development and Assessment Process (EMDAP), which is documented in RG 1.203. While RG 1.203 expands upon RG 1.157, there are certain aspects of RG 1.157 which are more detailed than RG 1.203; therefore, both RGs were used for the development of the FSLOCA EM.

2.2 WCOBRA/TRAC-TF2 Computer Code

The FSLOCA EM (Reference 1) uses the WCOBRA/TRAC-TF2 code to analyze the system thermal-hydraulic response for the full spectrum of break sizes. WCOBRA/TRAC-TF2 was created by combining a 1D module (TRAC-P) with a 3D module (based on Westinghouse modified COBRA-TF). The 1D and 3D modules include an explicit non-condensable gas transport equation. The use of TRAC-P allows for the extension of a two-fluid, six-equation formulation of the two-phase flow to the 1D loop components. This new code is WCOBRA/TRAC-TF2, where "TF2" is an identifier that reflects the use of a three-field

(TF) formulation of the 3D module derived by COBRA-TF and a two-fluid (TF) formulation of the 1D module based on TRAC-P.

This best-estimate computer code contains the following features:

1. Ability to model transient three-dimensional flows in different geometries inside the reactor vessel
2. Ability to model thermal and mechanical non-equilibrium between phases
3. Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes
4. Ability to represent important reactor components such as fuel rods, steam generators (SGs), reactor coolant pumps (RCPs), etc.

A detailed assessment of the computer code WCOBRA/TRAC-TF2 was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the ability of the code to predict key physical phenomena for a LOCA. Modeling of a LOCA introduces additional uncertainties which are identified and quantified in the plant-specific analysis. The reactor vessel and loop noding scheme used in the FSLOCA EM is consistent with the noding scheme used for the experiment simulations that form the validation basis for the physical models in the code. Such noding choices have been justified by assessing the model against large and full scale experiments.

2.3 Compliance with FSLOCA EM Limitations and Conditions

The NRC's SER for Reference 1 contains 15 limitations and conditions on the NRC-approved FSLOCA EM. A summary of each limitation and condition and how it was met is provided below.

<u>Limitation and Condition Number 1</u>

<u>Summary</u>

The FSLOCA EM is not approved to demonstrate compliance with 10 CFR 50.46 acceptance criterion (b)(5) related to the long-term cooling.

<u>Compliance</u>

The analysis for Watts Bar Units 1 and 2 with the FSLOCA EM is only being used to demonstrate compliance with 10 CFR 50.46 (b)(1) through (b)(4).

<u>Limitation and Condition Number 2</u>

<u>Summary</u>

The FSLOCA EM is approved for the analysis of Westinghouse-designed 3-loop and 4-loop PWRs with cold-side injection. Analyses should be executed consistent with the approved method, or any deviations from the approved method should be described and justified.

Compliance

Watts Bar Units 1 and 2 are Westinghouse-designed 4-loop PWRs with cold-side injection, so they are within the NRC-approved methodology. The analysis for Watts Bar Units 1 and 2 utilizes the NRC-approved FSLOCA methodology, except for the changes which were previously transmitted to the NRC pursuant to 10 CFR 50.46 in LTR-NRC-18-30 (Reference 5).

After completion of the analysis for Watts Bar Units 1 and 2, two errors were discovered in the WCOBRA/TRAC-TF2 code that can occur under certain conditions. These errors were found to have a negligible impact on analysis results with the FSLOCA EM as described in Reference 14.

The treatment for the uncertainty in the gamma energy redistribution is discussed on pages 29-75 and 29-76 of WCAP-16996-P-A, Revision 1 (Reference 1), and the equation for the assumed increase in hot rod and hot assembly relative power is presented on page 29-76. The power increase in the hot rod and hot assembly due to energy redistribution in the application of the FSLOCA EM to Watts Bar Units 1 and 2 was calculated incorrectly. This error resulted in a 0% to 5% deficiency in the modeled hot rod and hot assembly rod linear heat rates on a run-specific basis, depending on the as-sampled value for the uncertainty. The effect of the error correction was evaluated against the application of the FSLOCA EM to Watts Bar Units 1 and 2.

The error correction has only a limited impact on the power modeled for a single assembly in the core. As such, there is a negligible impact of the error correction on the system thermal-hydraulic response during the postulated LOCA.

For Region I, the primary impact of the error correction is on the rate of cladding heatup above the two-phase mixture level in the core during the boiloff phase. The PCT impact was assessed using run-specific PCT versus linear heat rate relationships and the run-specific hot rod and hot assembly linear heat rate increase that would result from the error correction. Using this approach, the correction of the error was estimated to increase the Region I analysis PCT by 2°F, leading to a final result of 978°F for the Region I analysis.

For Region II, parametric PWR sensitivity studies, derived from a subset of uncertainty analysis simulations covering various design features and fuel arrays, were examined to determine the sensitivity of the analysis results to the error correction. The PCT impact from the error correction was found to be different for the different transient phases (i.e., blowdown versus reflood) based on the PWR sensitivity studies and existing power distribution sensitivity studies. Based on the results from the PWR sensitivity studies, the correction of the error is estimated to increase the Region II analysis PCT by 20°F, leading to an analysis result of 1477°F for the Region II analysis assuming loss-of-offsite power and 1464°F for the Region II analysis assuming offsite power available.

All of the analysis results including the error correction continue to maintain compliance with the 10 CFR 50.46 acceptance criteria.

Limitation and Condition Number 3

Summary

For Region II, the containment pressure calculation will be executed in a manner consistent with the approved methodology (i.e., the COCO or LOTIC2 model will be based on appropriate plant-specific design parameters and conditions, and engineered safety features which can reduce pressure are modeled). This includes utilizing a plant-specific initial containment temperature, and only taking credit for containment coatings which are qualified and outside of the break zone-of-influence.

Compliance

The containment pressure calculation for the Watts Bar Units 1 and 2 analysis was performed consistent with the NRC-approved methodology. Appropriate design parameters and conditions were modeled, as were the engineered safety features which can reduce the containment pressure. A plant-specific initial temperature associated with normal full-power operating conditions was modeled, and no coatings were credited on any of the containment structures.

Limitation and Condition Number 4

Summary

The decay heat uncertainty multiplier will be [ ]^{a,c} The analysis simulations for the FSLOCA EM will not be executed for longer than 10,000 seconds following reactor trip unless the decay heat model is appropriately justified. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results will be provided in the analysis submittal in units of sigma and absolute units.

Compliance

Consistent with the NRC-approved methodology, the decay heat uncertainty multiplier was [ ]^{a,c} for the Watts Bar Units 1 and 2 analysis. The analysis simulations were all executed for no longer than 10,000 seconds following reactor trip. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results have been provided in units of sigma and approximate absolute units in Table 7.

Limitation and Condition Number 5

Summary

The maximum assembly and rod length-average burnup is limited to [ ]^{a,c} respectively.

Compliance

The maximum analyzed assembly and rod length-average burnup is less than or equal to [ ]^{a,c} respectively, for Watts Bar Units 1 and 2.

Limitation and Condition Number 6

Summary

The fuel performance data for analyses with the FSLOCA EM should be based on the PAD5 code (at present), which includes the effect of thermal conductivity degradation. The nominal fuel pellet average temperatures and rod internal pressures should be the maximum values, and the generation of all the PAD5 fuel performance data should adhere to the NRC-approved PAD5 methodology.

Compliance

PAD5 fuel performance data is utilized in the Watts Bar Units 1 and 2 analysis with the FSLOCA EM. The analyzed fuel pellet average temperatures bound the maximum values calculated in accordance with Section 7.5.1 of Reference 2, and the analyzed rod internal pressures were calculated in accordance with Section 7.5.2 of Reference 2.

Limitation and Condition Number 7

Summary

The YDRAG uncertainty parameter should be **[**

]^{a,c}

Compliance

Consistent with the NRC-approved methodology, the YDRAG uncertainty parameter was **[**

]^{a,c} for the Watts Bar Units 1 and 2 Region I analysis.

Limitation and Condition Number 8

Summary

The **[**

]^{a,c}

Compliance

Consistent with the NRC-approved methodology, the **[**

]^{a,c} for the Watts Bar Units 1 and 2 Region I analysis.

Limitation and Condition Number 9

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to confirm that the **[**
]^{a,c} for the plant design being analyzed. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Compliance

Watts Bar Units 1 and 2 are Westinghouse-designed 4-loop PWRs. The requested sensitivity study was performed for a 4-loop Westinghouse-designed PWR and is discussed in Reference 13.

Limitation and Condition Number 10

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to: 1) demonstrate that no unexplained behavior occurs in the predicted safety criteria across the region boundary, and 2) ensure that the **[** **]**^{a,c} must cover the equivalent 2 to 4-inch break range using RCS-volume scaling relative to the demonstration plant. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Additionally, the minimum sampled break area for the analysis of Region II should be 1 ft^2.

Compliance

Watts Bar Units 1 and 2 are Westinghouse-designed 4-loop PWRs. The requested sensitivity study was performed for a 4-loop Westinghouse-designed PWR and is discussed in Reference 13.

The minimum sampled break area for the Watts Bar Units 1 and 2 Region II analysis was 1 ft^2.

Limitation and Condition Number 11

Summary

There are various aspects of this Limitation and Condition, which are summarized below:

1. The **[** **]**^{a,c} the Region I and Region II analysis seeds, and the analysis inputs will be declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[** **]**^{a,c} and the Region I and Region II analysis seeds will not be changed throughout the remainder of the analysis once they have been declared and documented.
2. If the analysis inputs are changed after they have been declared and documented, for the intended purpose of demonstrating compliance with the applicable acceptance criteria, then the changes and associated rationale for the changes will be provided in the analysis submittal. Additionally, the preliminary values for peak cladding temperature (PCT), maximum local oxidation (MLO), and core-wide oxidation (CWO) which caused the input changes will be provided. These preliminary values are not subject to Appendix B verification, and archival of the supporting information for these preliminary values is not required.
3. Plant operating ranges which are sampled within the uncertainty analysis will be provided in the analysis submittal for both regions.

Compliance

This Limitation and Condition was met for the Watts Bar Units 1 and 2 analysis as follows:

1. The **[** **]**^{a,c} the Region I and Region II analysis seeds, and the analysis inputs were declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[** **]**^{a,c} and the Region I and Region II analyses seeds were not changed once they were declared and documented.
2. The analysis inputs were not changed once they were declared and documented.

3. The plant operating ranges which were sampled within the uncertainty analyses are provided for Watts Bar Units 1 and 2 in Table 1.

Limitation and Condition Number 12

Summary

The plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves must be adequately accounted for in analysis with the FSLOCA EM.

Compliance

A bounding plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves was modeled in the Watts Bar Units 1 and 2 analysis.

Limitation and Condition Number 13

Summary

In plant-specific models for analysis with the FSLOCA EM: 1) the **[** **]**^{a,c} and 2) the **[** **]**^{a,c}

Compliance

The **[** **]**^{a,c} in the analysis for Watts Bar Units 1 and 2. The **[** **]**^{a,c} in the analysis.

Limitation and Condition Number 14

Summary

For analyses with the FSLOCA EM to demonstrate compliance against the current 10 CFR 50.46 oxidation criterion, the transient time-at-temperature will be converted to an equivalent cladding reacted (ECR) using either the Baker-Just or the Cathcart-Pawel correlation. In either case, the pre-transient corrosion will be summed with the LOCA transient oxidation. If the Cathcart-Pawel correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 13 percent limit. If the Baker-Just correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 17 percent limit.

Compliance

For the Watts Bar Units 1 and 2 analysis, the Baker-Just correlation was used to convert the LOCA transient time-at-temperature to an ECR. The resulting LOCA transient ECR was then summed with the pre-existing corrosion for comparison against the 10 CFR 50.46 local oxidation acceptance criterion of 17%.

Limitation and Condition Number 15

Summary

The Region II analysis will be executed twice; once assuming loss-of-offsite power (LOOP) and once assuming offsite power available (OPA). The results from both analysis executions should be shown to be in compliance with the 10 CFR 50.46 acceptance criteria.

The **[** **]**^{a,c}

Compliance

The Region II uncertainty analysis for Watts Bar Units 1 and 2 was performed twice; once assuming a LOOP and once assuming OPA. The results from both analyses that were performed are in compliance with the 10 CFR 50.46 acceptance criteria (see Section 5.0).

The **[**
]^{a,c}

3.0 REGION I ANALYSIS

3.1 Description of Representative Transient

The small break LOCA transient can be divided into time periods in which specific phenomena are occurring, as discussed below.

Blowdown

The rapid depressurization of the RCS coincides with subcooled liquid flow through the break. Following the reactor trip on the low pressurizer pressure setpoint, the pressurizer drains, and safety injection is initiated on the low pressurizer pressure SI setpoint. After reaching this setpoint and applying the safety injection delays, high pressure safety injection (charging and high head SI pumps) flow begins. Phase separation begins in the upper head and upper plenum near the end of this period until the entire RCS approaches saturation, ending the rapid depressurization slightly above the steam generator secondary side pressure near the modeled main steam safety valve (MSSV) setpoint.

Natural Circulation

This quasi-equilibrium phase persists while the RCS pressure remains slightly above the secondary side pressure. The system drains from the top down, and while significant mass is continually lost through the break, the vapor generated in the core is trapped in the upper regions by the liquid remaining in the crossover leg loop seals. Throughout this period, the core remains covered by a two-phase mixture and the fuel cladding temperatures remain at the saturation temperature level.

Loop Seal Clearance

As the system drains, the liquid levels in the downhill side of the pump suction (crossover leg) become depressed all the way to the bottom elevations of the piping, allowing the steam trapped during the natural circulation phase to vent to the break (i.e., a process called loop seal clearance). The break flow and the flow through the RCS loops become primarily vapor. Relief of a static head imbalance allows for a quick but temporary recovery of liquid levels in the inner portion of the reactor vessel.

Boil-Off

With a vapor vent path established after the loop seal clearance, the RCS depressurizes at a rate controlled by the critical flow, which continues to be a primarily high quality mixture of water and steam. The RCS pressure remains high enough such that safety injection flow cannot make up for the primary system fluid inventory lost through the break, leading to core uncovery and a fuel rod cladding temperature heatup.

Core Recovery

The RCS pressure continues to decrease, and once it reaches that of the accumulator gas pressure, the introduction of additional ECCS water from the accumulators replenishes the reactor vessel inventory and recovers the core mixture level. The transient is considered over as the break flow is compensated by the injected flow.

3.2 Analysis Results

The Watts Bar Units 1 and 2 Region I analysis was performed in accordance with the NRC-approved methodology in Reference 1 with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The transient that produced the analysis PCT result is a cold leg break with a break diameter of 4.2-inches. The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Table 1. Control rod drop is modeled for breaks less than 1 square foot assuming a 2 second reactor trip signal delay time and a 3 second rod drop time. RCP trip is modeled coincident with reactor trip on the low pressurizer pressure setpoint for LOOP transients. When the low pressurizer pressure SI setpoint is reached, there is a delay to account for emergency diesel generator start-up, filling headers, etc., after which safety injection is initiated into the RCS.

The results of the Watts Bar Units 1 and 2 Region I uncertainty analysis are summarized in Table 4. The sampled decay heat multipliers for the Region I analysis cases are provided in Table 7.

Table 5 contains a sequence of events for the transient that produced the Region I analysis PCT result. Figures 1 through 13 illustrate the calculated key transient response parameters for this transient.

4.0 REGION II ANALYSIS

4.1 Description of Representative Transient

A large-break LOCA transient can be divided into phases in which specific phenomena are occurring. A convenient way to divide the transient is in terms of the various heatup and cooldown phases that the fuel assemblies undergo. For each of these phases, specific phenomena and heat transfer regimes are important, as discussed below.

Blowdown – Critical Heat Flux (CHF) Phase

In this phase, the break flow is initially subcooled, the discharge rate of coolant from the break is high, the core flow reverses, the fuel rods go through departure from nucleate boiling (DNB), and the cladding rapidly heats up and the reactor is shut down due to the core voiding. Control rod drop is not credited for breaks greater than 1 square foot in area.

The regions of the RCS with the highest initial temperatures (upper core, upper plenum, and hot legs) begin to flash during this period. This phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture level swells and a saturated mixture is pushed into the core by the intact loop RCPs, still rotating in single-phase liquid. As the fluid in the cold leg reaches saturation conditions, the discharge flow rate at the break decreases significantly.

Blowdown – Upward Core Flow Phase

Heat transfer is increased as the two-phase mixture is pushed into the core. The break discharge rate is reduced because the fluid becomes saturated at the break. This phase ends as the lower plenum mass is largely depleted, the fluid in the loops become two-phase, and the RCP head degrades.

Blowdown – Downward Core Flow Phase

The break flow begins to dominate and pulls flow down through the core as the RCP head degrades due to increased voiding, while liquid and entrained liquid flows also provide core cooling. Heat transfer in this period may be enhanced by liquid flow from the upper head. Once the system has depressurized to less than the accumulator cover pressure, the accumulators begin to inject cold water into the cold legs. During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is bypassed around the downcomer and sent out through the break. As the system pressure continues to decrease, the break flow and consequently the downward core flow are reduced. The system pressure approaches the containment pressure at the end of this last period of the blowdown phase.

During this phase, the core begins to heat up as the system approaches containment pressure, and the phase ends when the reactor vessel begins to refill with ECCS water.

Refill Phase

The core continues to heat up as the lower plenum refills with ECCS water. This phase is characterized by a rapid increase in fuel cladding temperature at all elevations due to the lack of liquid and steam flow in the core region. The water completely refills the lower plenum and the refill phase ends. As ECCS water enters the core, the fuel rods in the lower core region begin to quench and liquid entrainment begins, resulting in increased fuel rod heat transfer.

Reflood Phase

During the early reflood phase, the accumulators begin to empty and nitrogen is discharged into the RCS. The nitrogen surge forces water into the core, which is then evaporated, causing system re-pressurization and a temporary reduction of pumped ECCS flow; this re-pressurization is illustrated by the increase in RCS pressure. During this time, core cooling may increase due to vapor generation and liquid entrainment, but conversely the early reflood pressure spike results in loss of mass out through the broken cold leg.

The pumped ECCS water aids in the filling of the downcomer throughout the reflood period. As the quench front progresses further into the core, the PCT elevation moves increasingly higher in the fuel assembly.

As the transient progresses, continued injection of pumped ECCS water refloods the core, effectively removes the reactor vessel metal mass stored energy and core decay heat, and leads to an increase in the reactor vessel fluid mass. Eventually the core inventory increases enough that liquid entrainment is able to quench all the fuel assemblies in the core.

4.2 Analysis Results

The Watts Bar Units 1 and 2 Region II analysis was performed in accordance with the NRC-approved methodology in Reference 1 with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The analysis was performed assuming both LOOP and OPA, and the results of both of the LOOP and OPA analyses are compared to the 10 CFR 50.46 acceptance criteria. The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Table 1. The results of the Watts Bar Units 1 and 2 Region II LOOP and OPA uncertainty analyses are summarized in Table 4. The sampled decay heat multipliers for the Region II analysis cases are provided in Table 7.

Table 6 contains a sequence of events for the transient that produced the more limiting analysis PCT result relative to the offsite power assumption. Figures 14 through 27 illustrate the key transient response parameters for this transient.

The containment pressure is calculated using the LOTIC2 code (References 11 and 12) for ice condenser containments. The assumed, conservatively low, containment pressure response used for the Watts Bar

Units 1 and 2 Region II analysis is compared to the calculated containment backpressure in Figure 21, consistent with the methodology in Reference 1.

5.0 COMPLIANCE WITH 10 CFR 50.46

It must be demonstrated that there is a high level of probability that the following criteria in 10 CFR 50.46 are met:

(b)(1) The analysis PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT is less than 2,200°F, the analysis with the FSLOCA EM confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Cladding Temperature less than 2,200°F," is demonstrated.

The results are shown in Table 4 for Watts Bar Units 1 and 2.

(b)(2) The analysis MLO corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting MLO is less than 17 percent when converting the time-at-temperature to an equivalent cladding reacted using the Baker-Just correlation and adding the pre-transient corrosion, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Maximum Local Oxidation of the cladding less than 17 percent," is demonstrated.

The results are shown in Table 4 for Watts Bar Units 1 and 2.

(b)(3) The analysis CWO corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. Since the resulting CWO is less than 1 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent," is demonstrated.

The results are shown in Table 4 for Watts Bar Units 1 and 2.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains in a coolable geometry.

This criterion is met by demonstrating compliance with criteria (b)(1), (b)(2), and (b)(3), and by assuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed. Criteria (b)(1), (b)(2), and (b)(3) have been met for Watts Bar Units 1 and 2 as shown in Table 4.

It is discussed in Section 32.1 of the NRC-approved FSLOCA EM (Reference 1) that the effects of LOCA and seismic loads on the core geometry do not need to be considered unless fuel assembly grid deformation extends beyond the core periphery (i.e., deformation in a fuel assembly with no sides adjacent to the core baffle plates). Inboard grid deformation due to combined LOCA and seismic loads is not calculated to occur for Watts Bar Units 1 and 2.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS.

Long-term cooling is dependent on the demonstration of the continued delivery of cooling water to the core. The actions that are currently in place to maintain long-term cooling are not impacted by the application of the NRC-approved FSLOCA EM (Reference 1).

Based on the analysis results for Region I and Region II presented in Table 4 for Watts Bar Units 1 and 2, it is concluded that Watts Bar Units 1 and 2 comply with the criteria in 10 CFR 50.46.

6.0 REFERENCES

1. "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," WCAP-16996-P-A, Revision 1, November 2016.
2. "Westinghouse Performance Analysis and Design Model (PAD5)," WCAP-17642-P-A, Revision 1, November 2017.
3. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974.
4. "Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," USNRC, August 1998.
5. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2017," LTR-NRC-18-30, July 2018.
6. "Emergency Core Cooling Systems: Revisions to Acceptance Criteria," Federal Register, V53, N180, pp. 35996-36005, September 1988.
7. "Best Estimate Calculations of Emergency Core Cooling System Performance," Regulatory Guide 1.157, USNRC, May 1989.
8. "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A Volume 1, Revision 2 and Volumes 2-5, Revision 1, March 1998.
9. "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," WCAP-16009-P-A, January 2005.
10. "Transient and Accident Analysis Methods," Regulatory Guide 1.203, USNRC, December 2005.
11. "Westinghouse Emergency Core Cooling System Evaluation Model – Summary," WCAP-8339, June 1974.
12. "Long Term Ice Condenser Containment Code – LOTIC Code," WCAP-8354-P-A, Supplement 1, April 1976.
13. "'Information to Satisfy the FULL SPECTRUM LOCA (FSLOCA) Evaluation Methodology Plant Type Limitations and Conditions for 4-loop Westinghouse Pressurized Water Reactors (PWRs)' (Proprietary/Non-Proprietary)," LTR-NRC-18-50, July 2018.
14. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2018," LTR-NRC-19-6, February 2019.

Table 1. Plant Operating Range Analyzed and Key Parameters for Watts Bar Units 1 and 2

Parameter		As-Analyzed Value or Range
1.0	**Core Parameters**	
	a) Core power	≤ 3479.8 MWt ± 0% Uncertainty
	b) Fuel type	17x17 RFA-2, ZIRLO®[1] Clad with IFMs
	c) Maximum total core peaking factor (F_Q), including uncertainties	2.5
	d) Maximum hot channel enthalpy rise factor ($F_{\Delta H}$), including uncertainties	1.65
	e) Axial flux difference (AFD) band at 100% power	-12% / +7%
	f) Maximum transient operation fraction	0.25
2.0	**Reactor Coolant System Parameters**	
	a) Thermal design flow (TDF)	93,100 gpm/loop
	b) Vessel average temperature (T_{AVG})	580.2°F ≤ T_{avg} ≤ 594.2°F
	c) Pressurizer pressure (P_{RCS})	2180 psia ≤ P_{RCS} ≤ 2300 psia
	d) Reactor coolant pump (RCP) model and power	Model 93A, 7000 hp
3.0	**Containment Parameters**	
	a) Containment modeling	Region I: Constant pressure equal to initial containment pressure Region II: Conservatively low containment pressure (Figure 21)
4.0	**Steam Generator (SG) and Secondary Side Parameters**	
	a) Steam generator tube plugging level	≤ 12%
	b) Main feedwater temperature	Nominal (441.8°F)
	c) Auxiliary feedwater temperature	Nominal (80°F)
	d) Auxiliary feedwater flow rate	165 gpm/SG

[1] ZIRLO is a registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

Table 1. Plant Operating Range Analyzed and Key Parameters for Watts Bar Units 1 and 2

Parameter		As-Analyzed Value or Range
5.0	**Safety Injection (SI) Parameters**	
	a) Single failure configuration	ECCS: Loss of one train of pumped ECCS Region II containment pressure: All SI trains are available
	b) Safety injection temperature (T_{SI})	60°F ≤ T_{SI} ≤ 105°F
	c) Low pressurizer pressure safety injection safety analysis limit	1715 psia
	d) Initiation delay time from low pressurizer pressure SI setpoint to full SI flow	≤ 40 seconds (OPA) or 55 seconds (LOOP)
	e) Safety injection flow	Minimum flows in Table 2a and Table 2b (Region I) or Table 3a and Table 3b (Region II)
6.0	**Accumulator Parameters**	
	a) Accumulator temperature (T_{ACC})	100°F ≤ T_{ACC} ≤ 130°F
	b) Accumulator water volume (V_{ACC})	1005 ft^3 ≤ V_{ACC} ≤ 1095 ft^3
	c) Accumulator pressure (P_{ACC})	585 psig ≤ P_{ACC} ≤ 690 psig
	d) Accumulator boron concentration	≥ 3000 ppm
7.0	**Reactor Protection System Parameters**	
	a) Low pressurizer pressure reactor trip signal processing time	≤ 2 seconds
	b) Low pressurizer pressure reactor trip setpoint	1715 psia
8.0	**Refueling Water Storage Tank (RWST) / Switchover Parameters**	
	a) Usable RWST volume	≥ 202,000 gallons
	b) Interruption time for switchover to cold leg recirculation	0 seconds
	c) SI flow after switchover to cold leg recirculation	Table 2b
	d) SI temperature after switchover to cold leg recirculation	≤ 150°F

Table 2a. Safety Injection Flow Used for Region I Calculation for Watts Bar Units 1 and 2

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Charging Flow (gpm)
14.7	413.7	260.6
114.7	399.4	251.9
214.7	383.7	243.1
314.7	366.2	234.2
414.7	348.8	225.1
514.7	330.9	216.0
614.7	311.2	206.3
714.7	290.5	196.4
814.7	270.4	186.2
914.7	248.7	175.8
1014.7	223.9	165.2
1114.7	196.8	154.0
1214.7	164.9	142.7
1314.7	120.0	128.2
1414.7	41.1	113.2
1514.7	0.0	96.9
1614.7	0.0	79.8
1714.7	0.0	60.6
1814.7	0.0	41.5
1914.7	0.0	24.8
2014.7	0.0	4.6
2114.7	0.0	0.0

Table 2b. Safety Injection Flow After Switchover to Cold Leg Recirculation Used for Region I Calculation for Watts Bar Units 1 and 2

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Charging Flow (gpm)
14.7	394.4	247.7
114.7	381.4	240.4
214.7	367.7	233.0
314.7	353.6	225.5
414.7	338.8	217.7
514.7	323.4	209.9
614.7	307.2	201.8
714.7	290.1	193.5
814.7	271.9	185.0
914.7	252.3	176.3
1014.7	230.9	167.3
1114.7	207.3	158.0
1214.7	180.7	148.3
1314.7	149.1	135.6
1414.7	108.4	120.6
1514.7	11.2	105.1
1614.7	0.0	88.8
1714.7	0.0	70.4
1814.7	0.0	49.5
1914.7	0.0	32.8
2014.7	0.0	14.6
2114.7	0.0	0.0

Table 3a. Safety Injection Flow Used for Region II Calculation for Watts Bar Units 1 and 2

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Charging Flow (gpm)
14.7	413.8	260.6
34.7	410.5	258.8
54.7	407.2	257.1
74.7	403.8	255.4
94.7	400.5	253.6
114.7	397.2	251.9
134.7	393.3	250.1
154.7	389.4	248.4
174.7	385.5	246.6
214.7	377.7	243.1
314.7	356.9	234.2
414.7	335.3	225.1
514.7	313.2	216.0
614.7	290.0	206.3
714.7	264.8	196.4
814.7	237.3	186.2
914.7	208.5	175.8
1014.7	178.3	165.2
1114.7	140.6	154.0
1214.7	93.0	142.7
1314.7	20.7	128.2
1414.7	0.0	113.2
1514.7	0.0	96.9
1614.7	0.0	79.8
1714.7	0.0	60.6
1814.7	0.0	41.5
1914.7	0.0	24.8
2014.7	0.0	4.6
2114.7	0.0	0.0

Table 3b. Safety Injection Flow Used for Region II Calculation for Watts Bar Units 1 and 2

Pressure (psia)	Low Head Safety Injection (LHSI) Flow (gpm)
14.7	2697.8
34.7	2264.0
54.7	1782.4
74.7	1349.6
94.7	1097.1
114.7	736.8
134.7	293.5
154.7	0.0

Table 4. Watts Bar Units 1 and 2 Analysis Results with the FSLOCA EM

Outcome	Region I Value	Region II Value (OPA)	Region II Value (LOOP)
95/95 PCT[1]	976°F + 2°F = 978°F	1,444°F + 20°F = 1,464°F	1,457°F + 20°F = 1,477°F
95/95 MLO[2]	8.9%	8.9%	8.9%
95/95 CWO	0.0%	0.0%	0.0%

Notes:

1. The PCT presented in the table shows the analysis-of-record result, which is the sum of the uncertainty analysis result plus the impact of the energy redistribution uncertainty error correction. The figures presenting the analysis results correspond to the uncertainty analysis result. The MLO and CWO were confirmed to maintain compliance with the 10 CFR 50.46 acceptance criteria with the error correction.
2. Due to the low amounts of predicted transient oxidation, the 95/95 MLO results are comprised of pre-transient oxidation only.

Table 5. Watts Bar Units 1 and 2 Sequence of Events for Region I Analysis PCT Transient

Event	Time after Break (sec)
Start of Transient	0
Reactor Trip Signal	23
Safety Injection Signal	23
Safety Injection Begins	78
Loop Seal Clearing Occurs	498
Top of Core Uncovered	700
Accumulator Injection Begins	866
PCT Occurs	880
Top of Core Recovered	962

Table 6. Watts Bar Units 1 and 2 Sequence of Events for Region II Analysis PCT Transient

Event	Time after Break (sec)
Start of Transient	0.0
Fuel Rod Burst Occurs	4.5
Safety Injection Signal	5.8
PCT Occurs	14.9
Accumulator Injection Begins	17.5
End of Blowdown	32.0
Safety Injection Begins	60.8
Accumulator Empty	72.5
All Rods Quenched	110

Table 7. Watts Bar Units 1 and 2 Sampled Value of Decay Heat Uncertainty Multiplier, DECAY_HT, for Region I and Region II Analysis Cases

Region	Case	DECAY_HT (units of σ)	DECAY_HT (absolute units)[1]
Region I	PCT	+ 1.1574	5.81%
	MLO	+ 0.1438	0.74%
	CWO[2]	N/A	N/A
Region II (OPA)	PCT	+ 1.3765	7.08%
	MLO	+ 0.4224	2.18%
	CWO[2]	N/A	N/A
Region II (LOOP)	PCT	+ 1.3765	7.08%
	MLO	+ 0.4224	2.18%
	CWO[2]	N/A	N/A

Notes:

1. Approximate uncertainty in total decay heat power at 1 second after shutdown as defined by the ANSI/ANS-5.1-1979 decay heat standard for ^{235}U, ^{239}Pu, and ^{238}U assuming infinite operation.
2. Since the 95/95 CWO outcome was 0.0% (See Table 4), which was reflective of numerous transient simulations, CWO results are not reported.

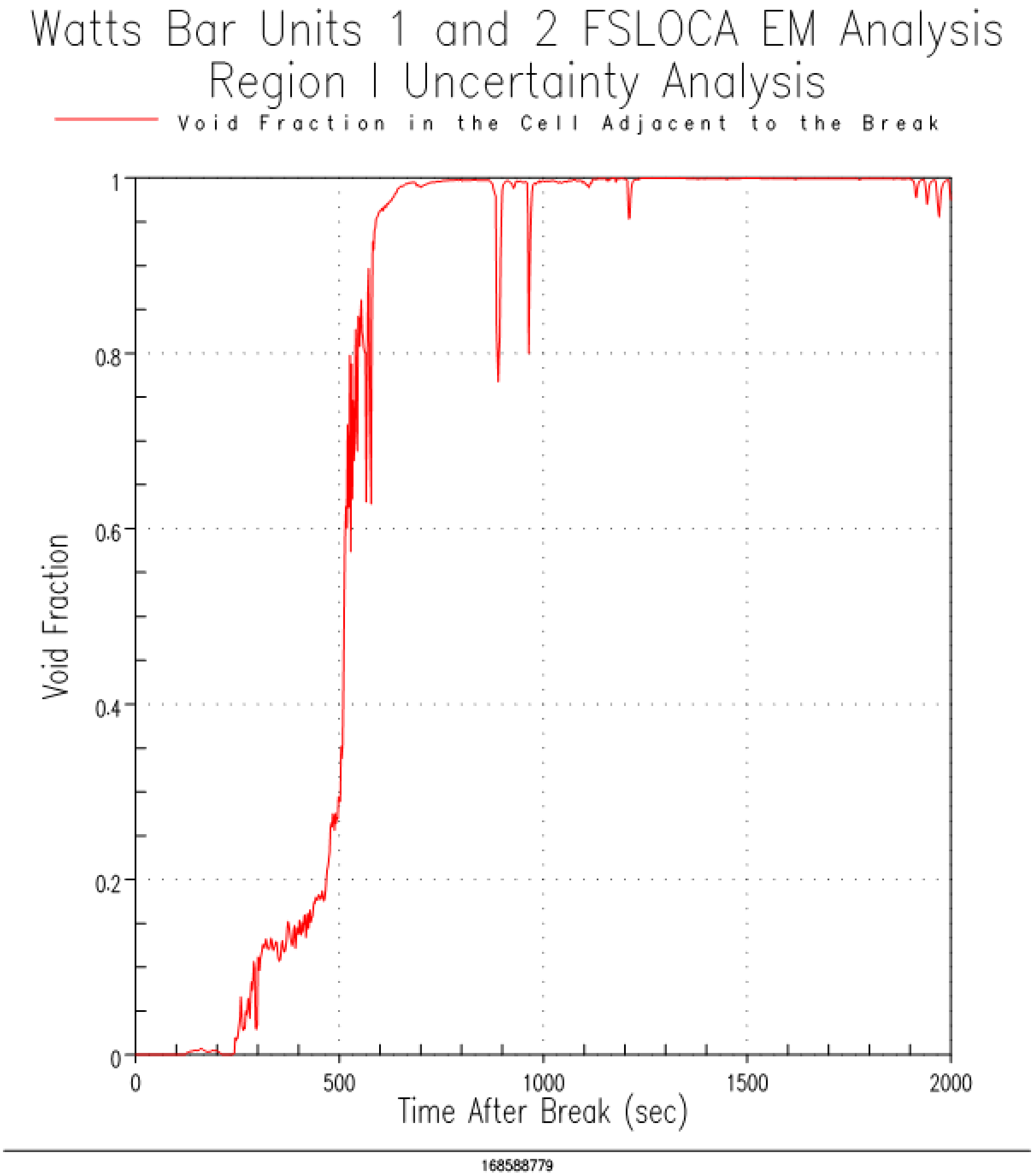

Figure 1: Watts Bar Units 1 and 2 Break Flow Void Fraction for Region I Analysis PCT Case

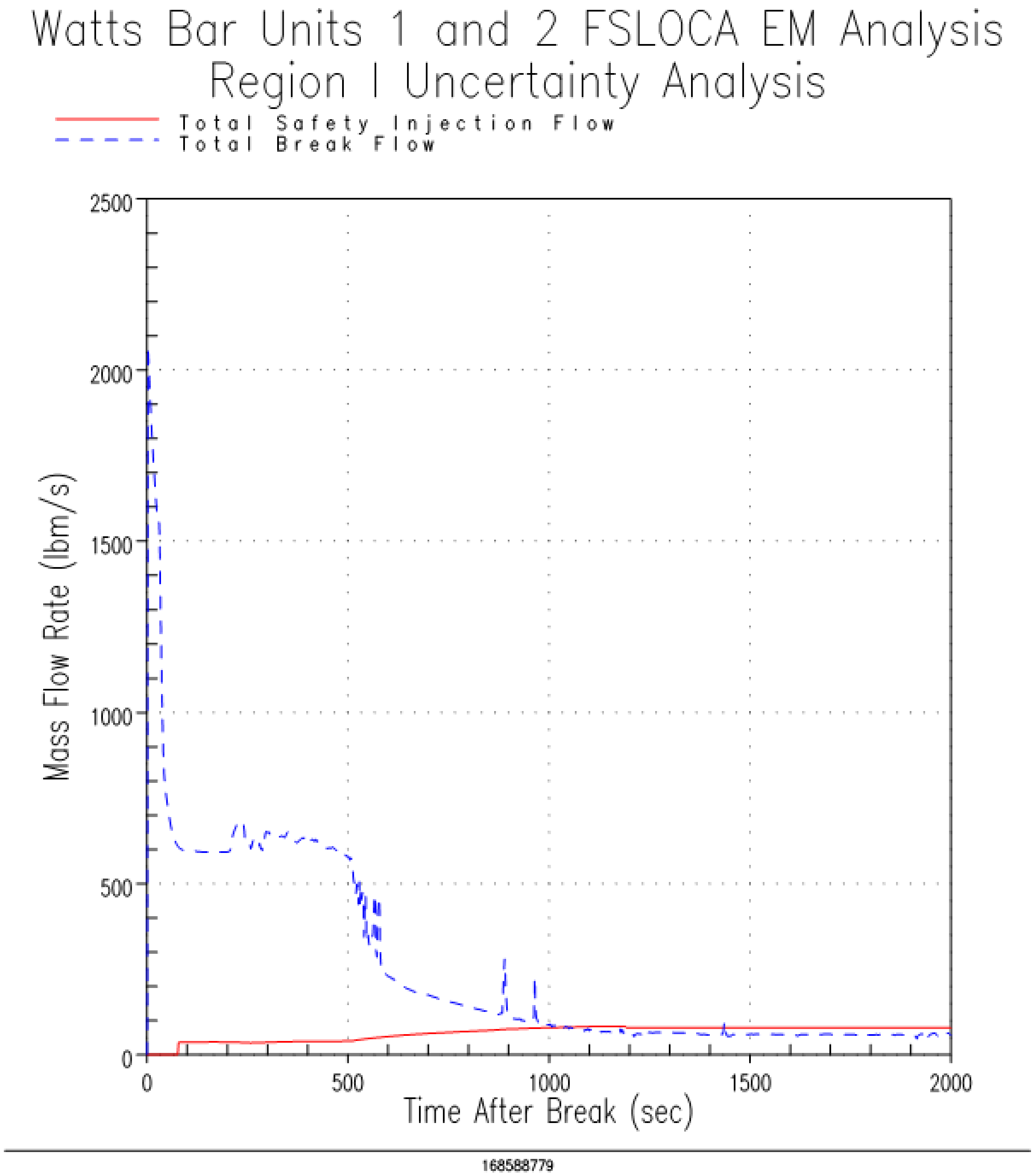

Figure 2: Watts Bar Units 1 and 2 Total Pumped SI Flow and Total Break Flow for Region I Analysis PCT Case

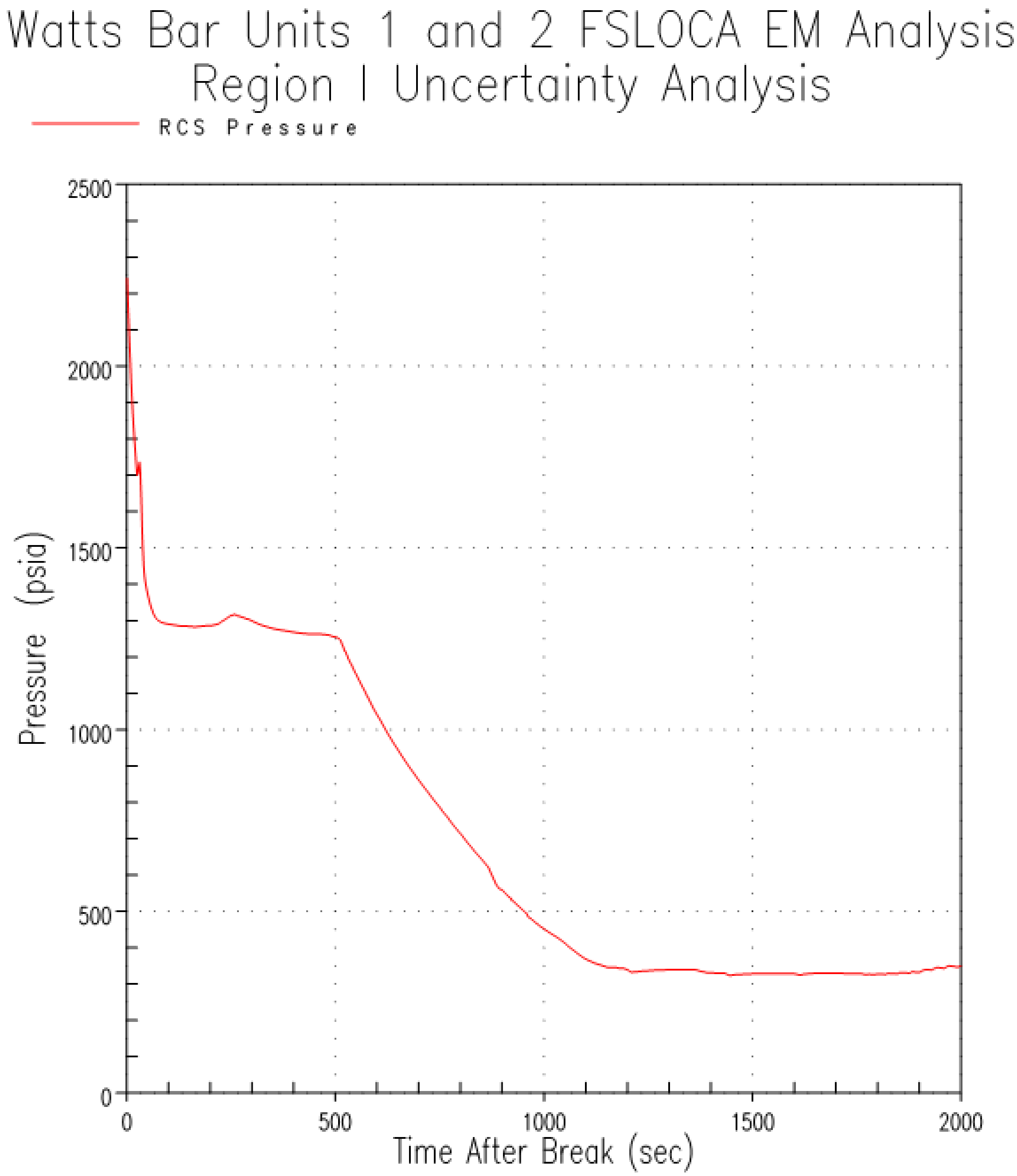

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Figure 3: Watts Bar Units 1 and 2 RCS Pressure for Region I Analysis PCT Case

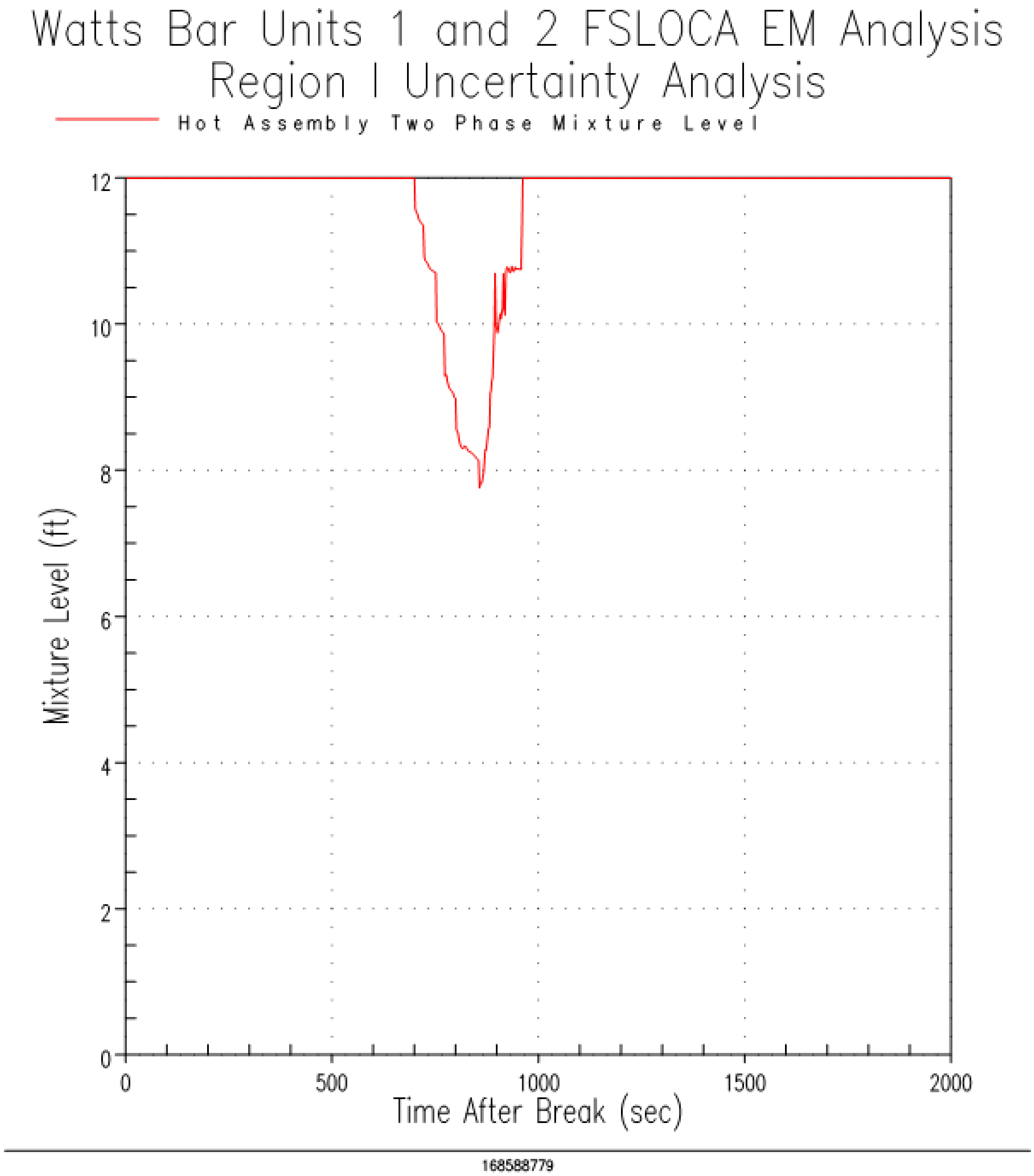

Figure 4: Watts Bar Units 1 and 2 Hot Assembly Two-Phase Mixture Level (where 0 ft is bottom of active fuel, 12 ft is top of active fuel) for Region I Analysis PCT Case

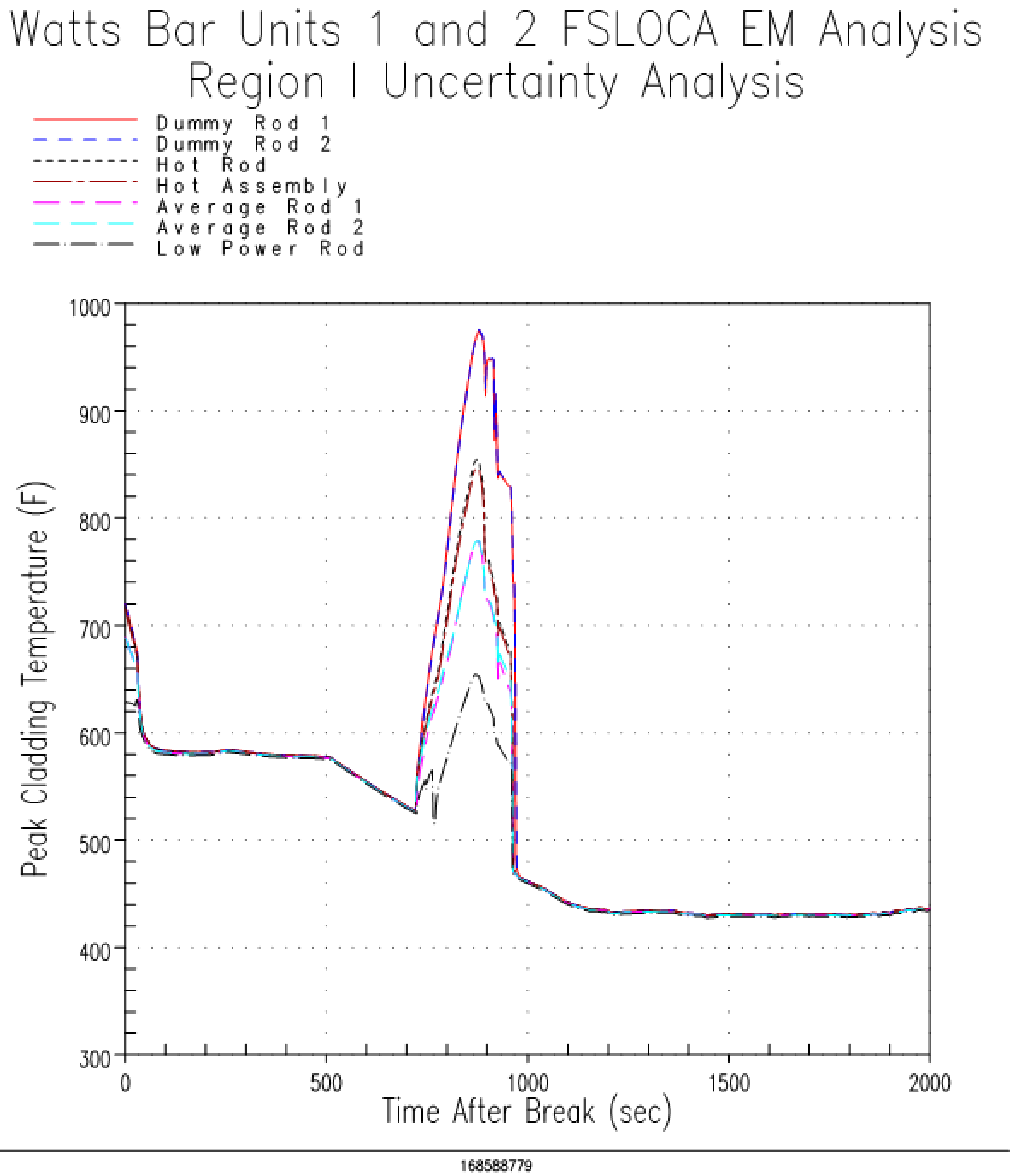

Figure 5: Watts Bar Units 1 and 2 Peak Cladding Temperature for all Rods for Region I Analysis PCT Case

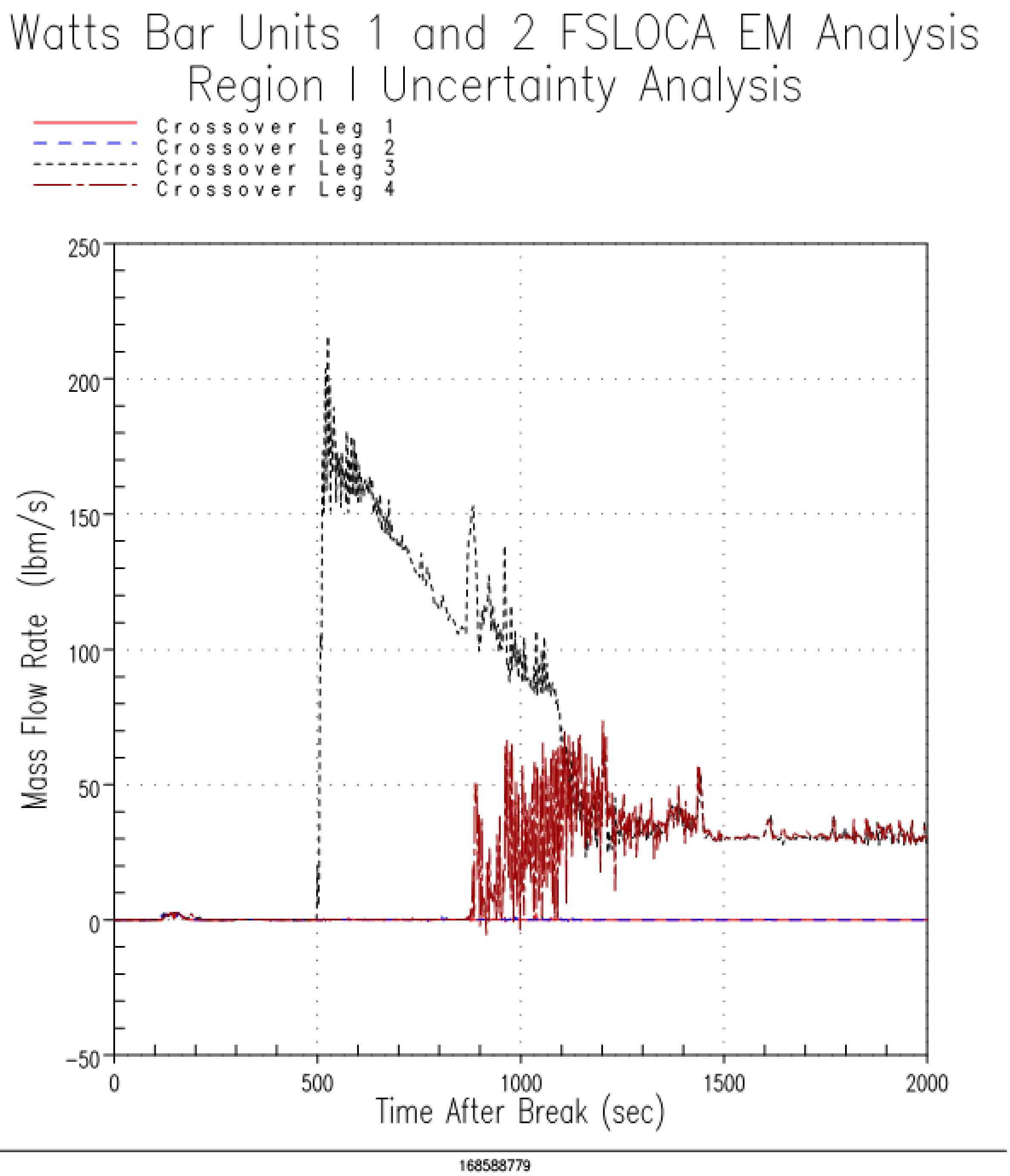

Figure 6: Watts Bar Units 1 and 2 Vapor Mass Flow Rate through the Crossover Legs for Region I Analysis PCT Case

Note: Loop 4 is the broken loop; Loop 2 & 3 flow overlay.

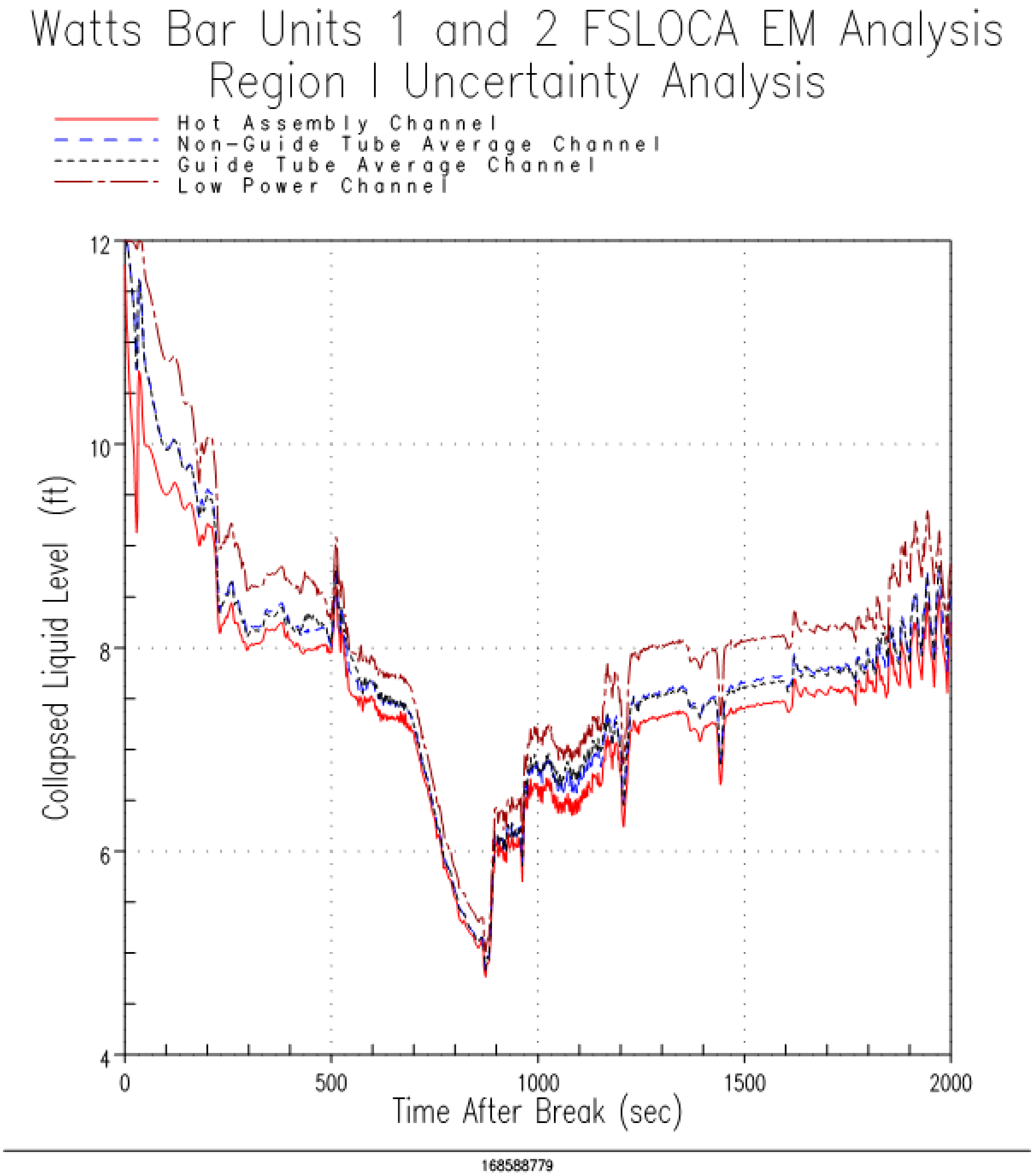

Figure 7: Watts Bar Units 1 and 2 Core Collapsed Liquid Levels (where 0 ft is bottom of active fuel, 12 ft is top of active fuel) for Region I Analysis PCT Case

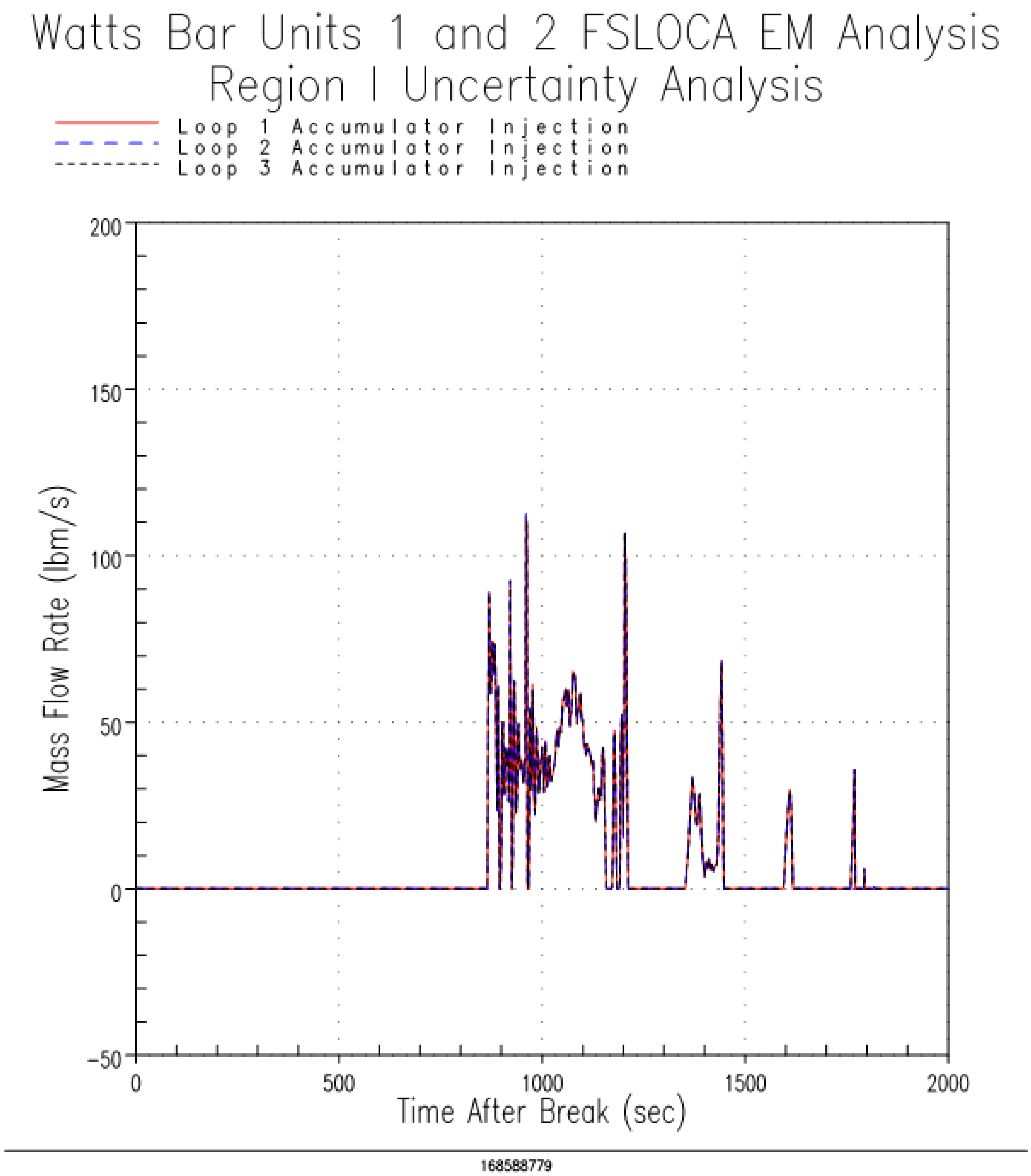

Figure 8: Watts Bar Units 1 and 2 Accumulator Injection Flow for Region I Analysis PCT Case

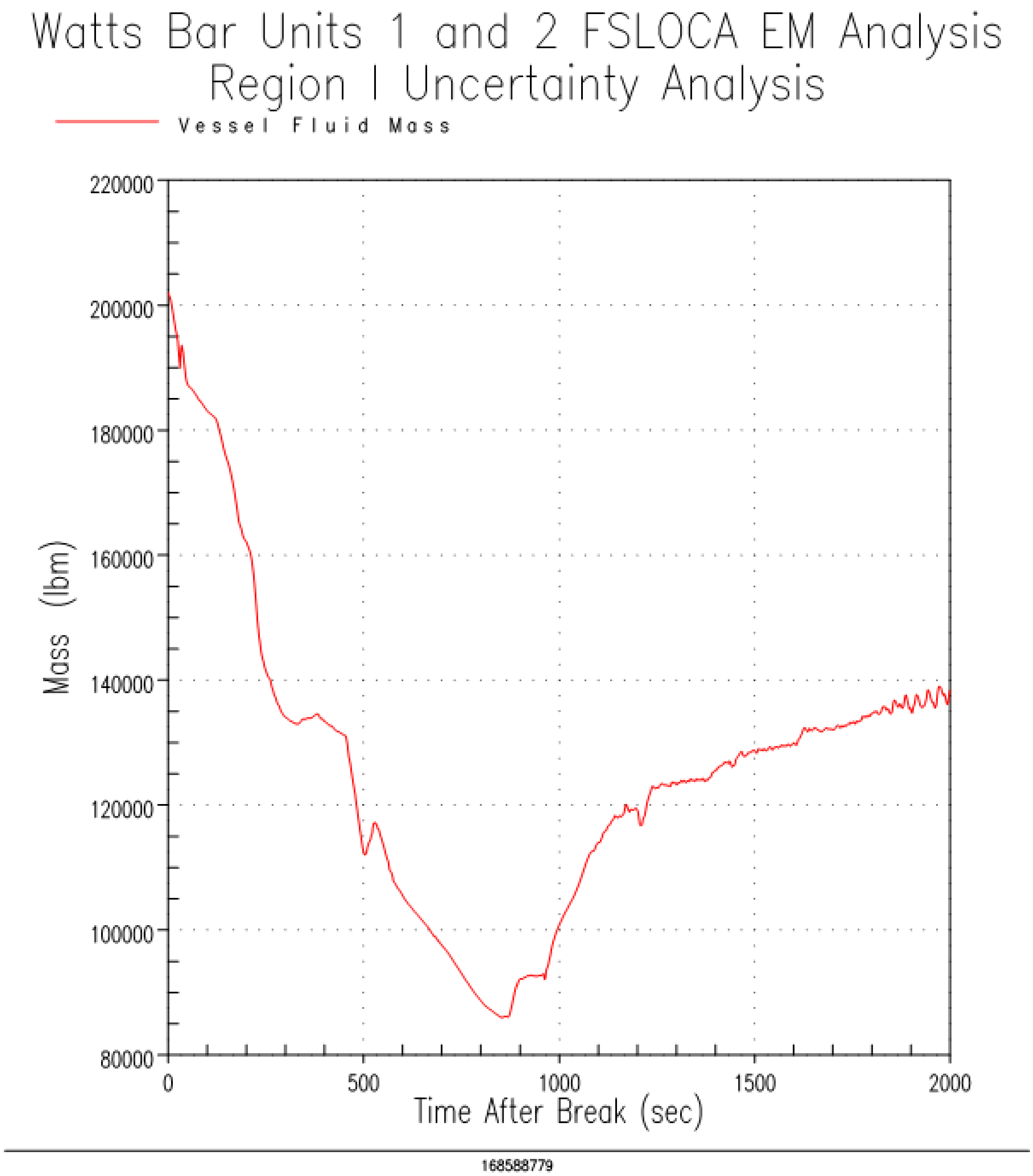

Figure 9: Watts Bar Units 1 and 2 Vessel Fluid Mass for Region I Analysis PCT Case

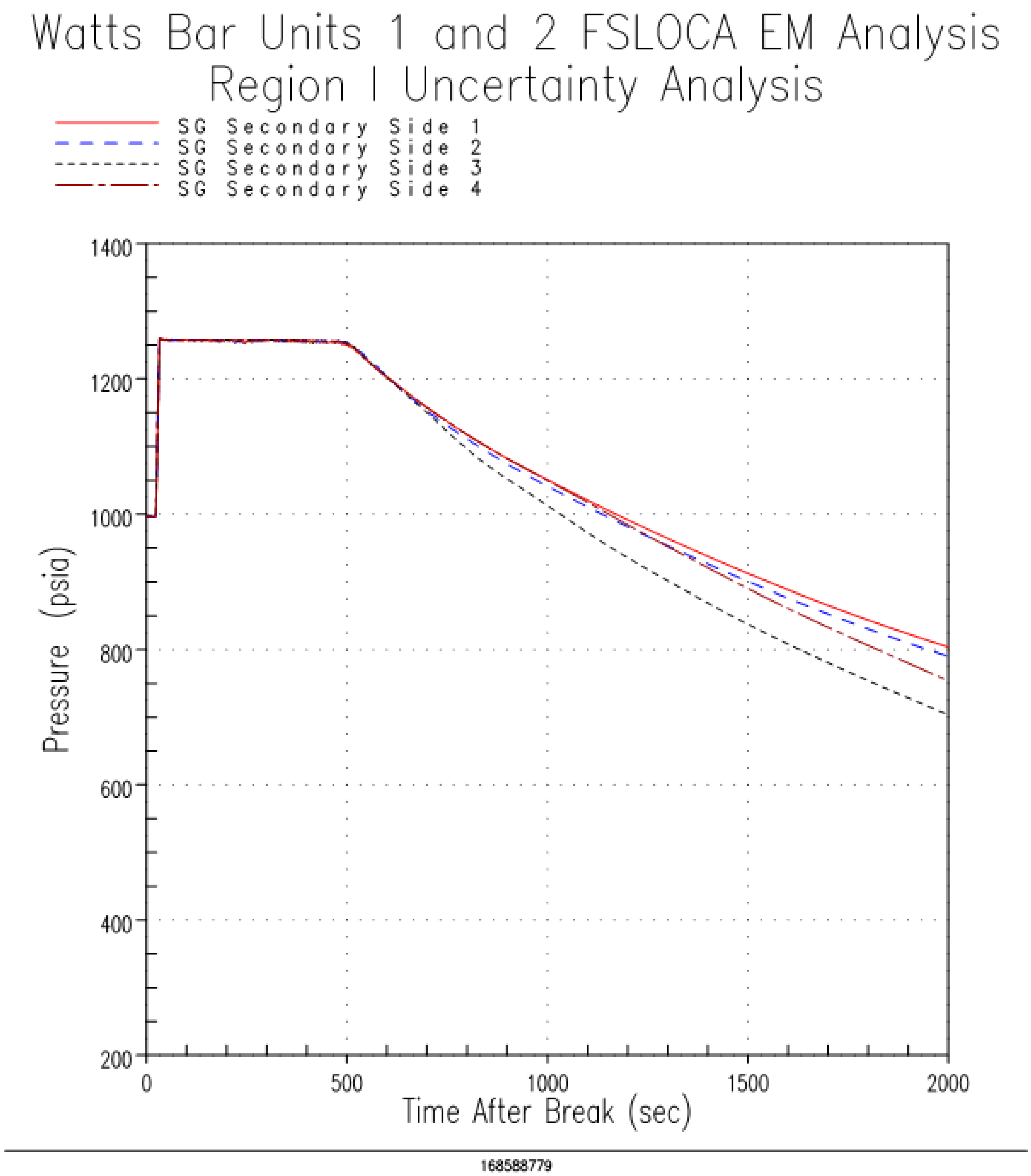

Figure 10: Watts Bar Units 1 and 2 Steam Generator Secondary Side Pressure for Region I Analysis PCT Case

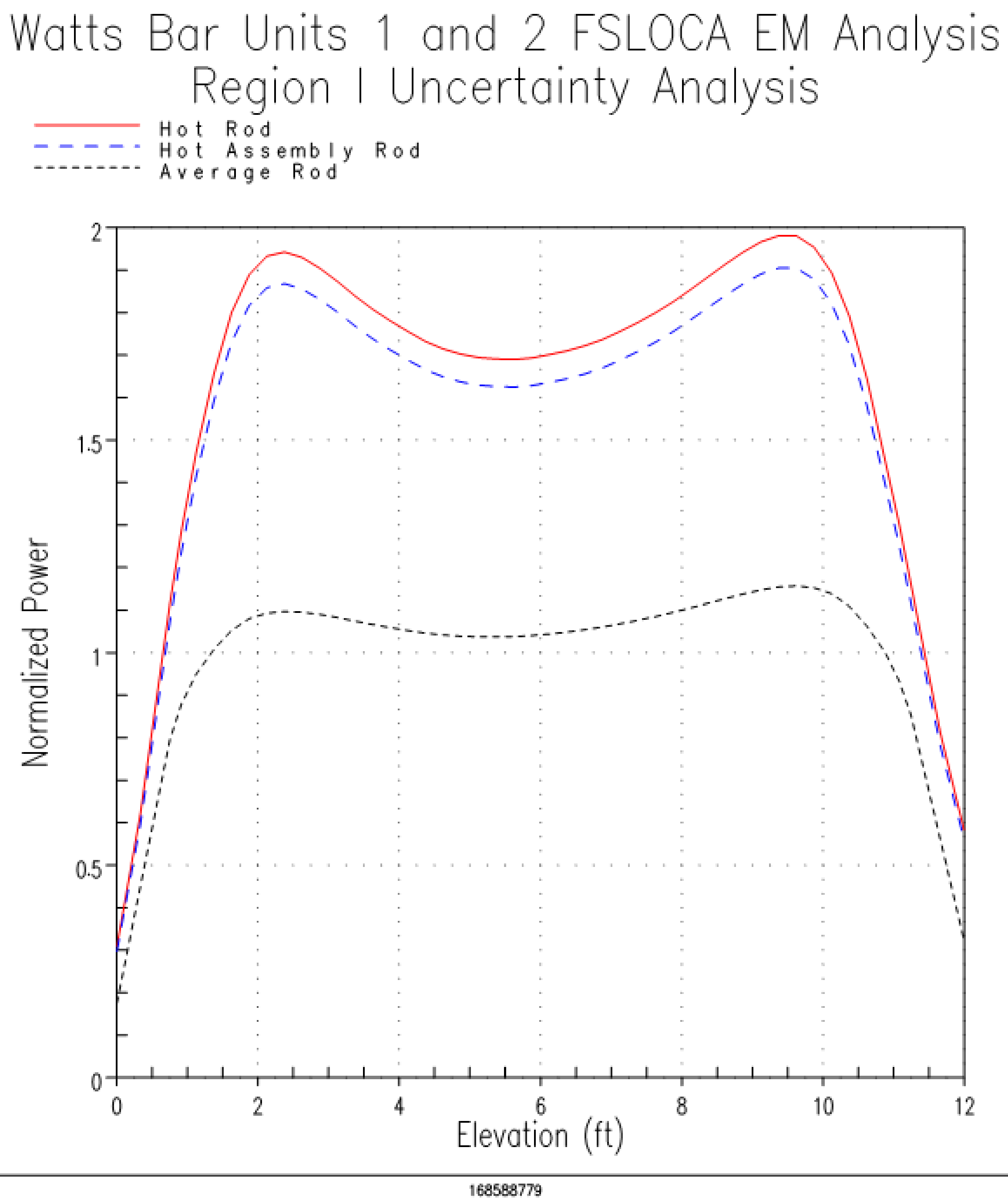

Figure 11: Watts Bar Units 1 and 2 Normalized Core Power Shapes for Region I Analysis PCT Case

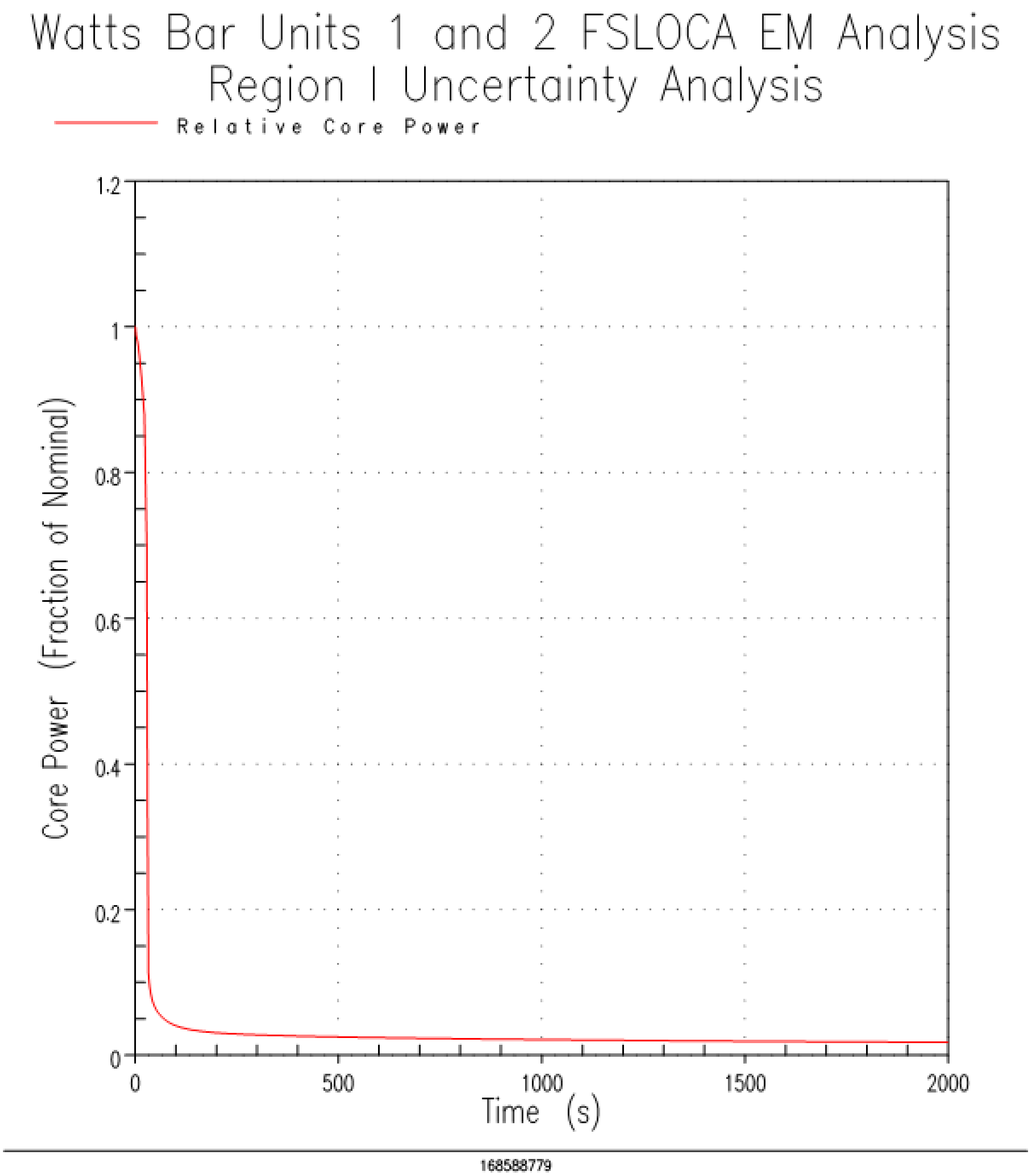

Figure 12: Watts Bar Units 1 and 2 Relative Core Power for Region I Analysis PCT Case

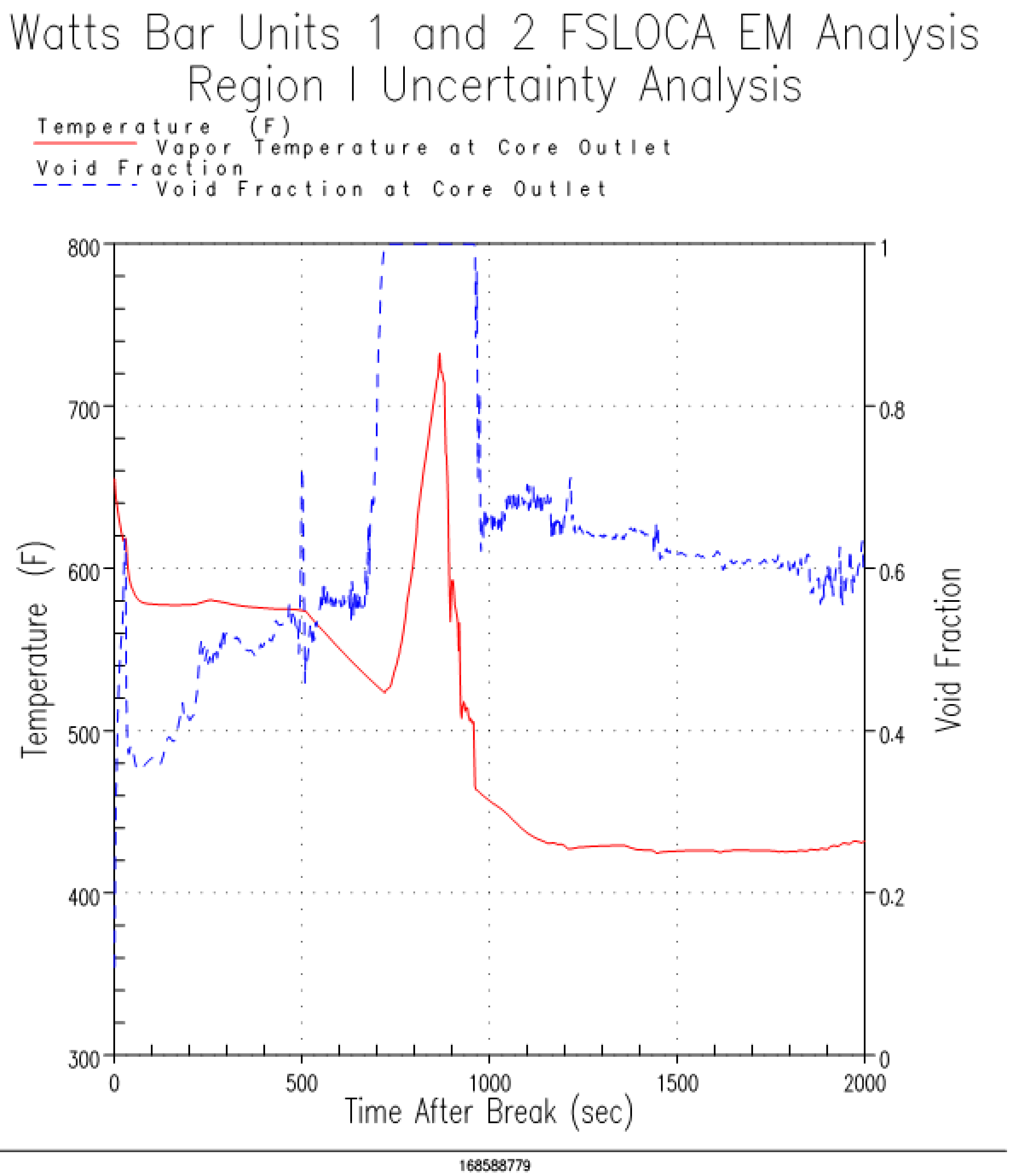

Figure 13: Watts Bar Units 1 and 2 Vapor Temperature and Void Fraction at Core Outlet for Region I Analysis PCT Case

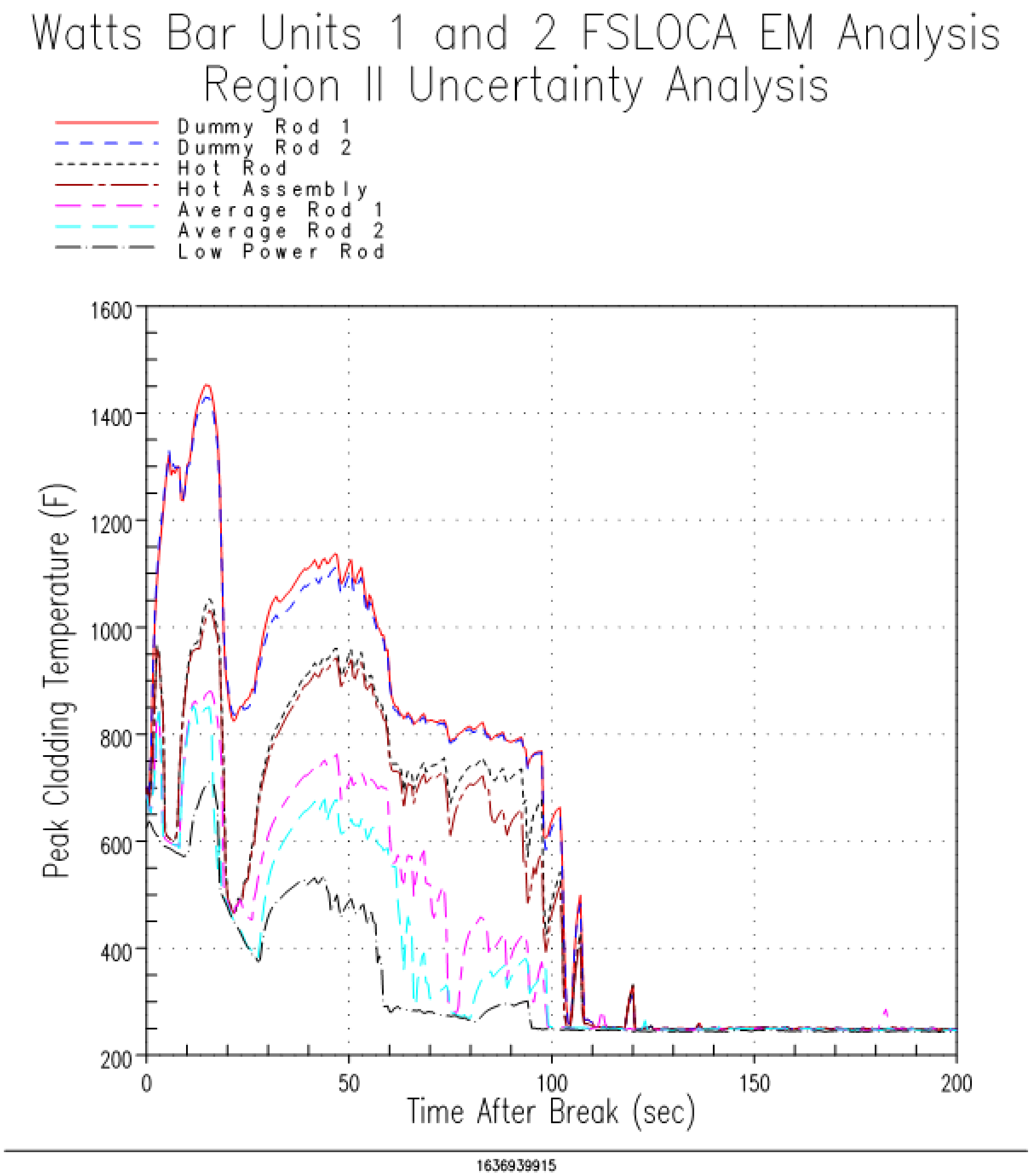

Figure 14: Watts Bar Units 1 and 2 Peak Cladding Temperature for all Rods for Region II Analysis PCT Case

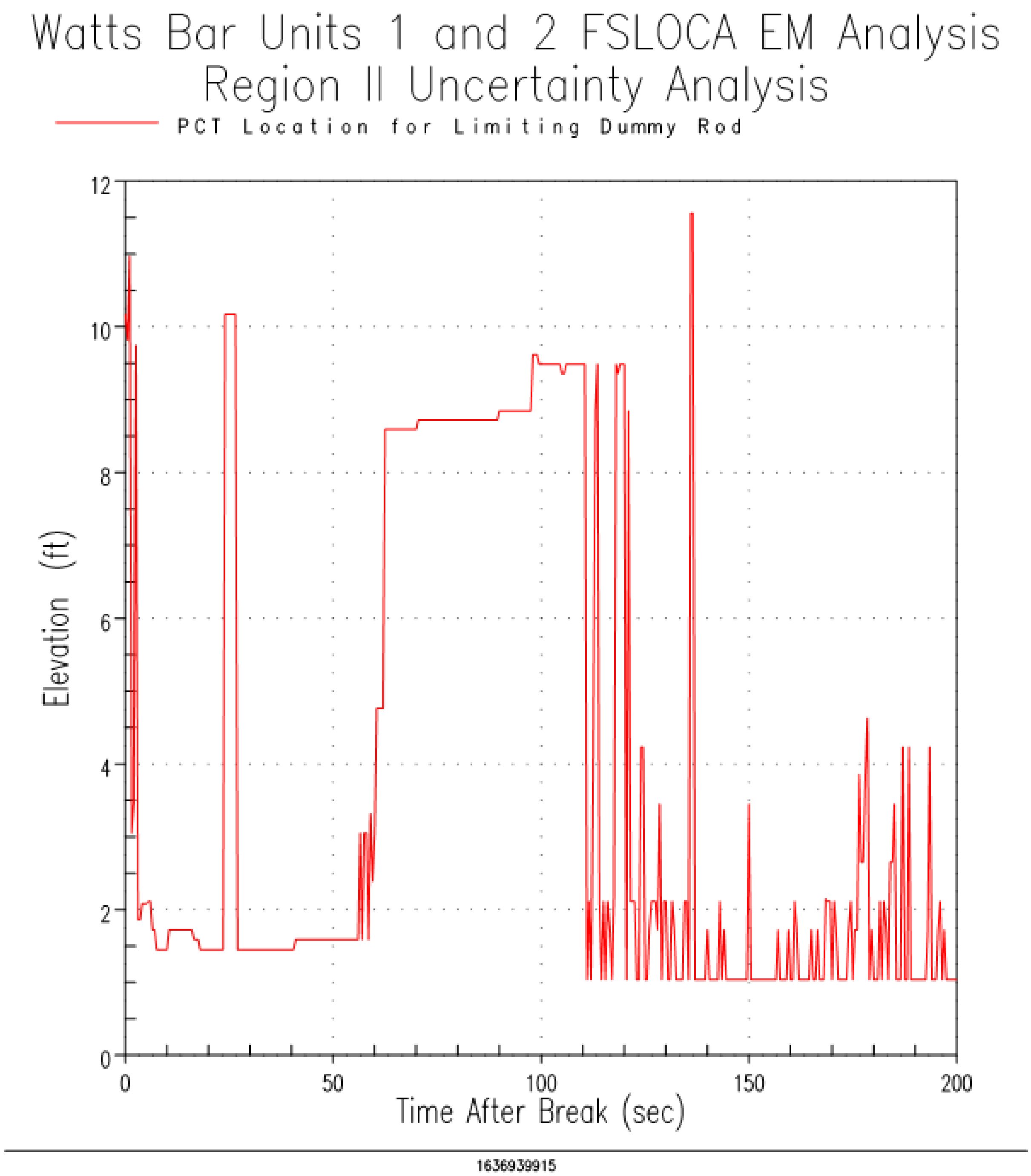

Figure 15: Watts Bar Units 1 and 2 Peak Cladding Temperature Elevation for Region II Analysis PCT Case

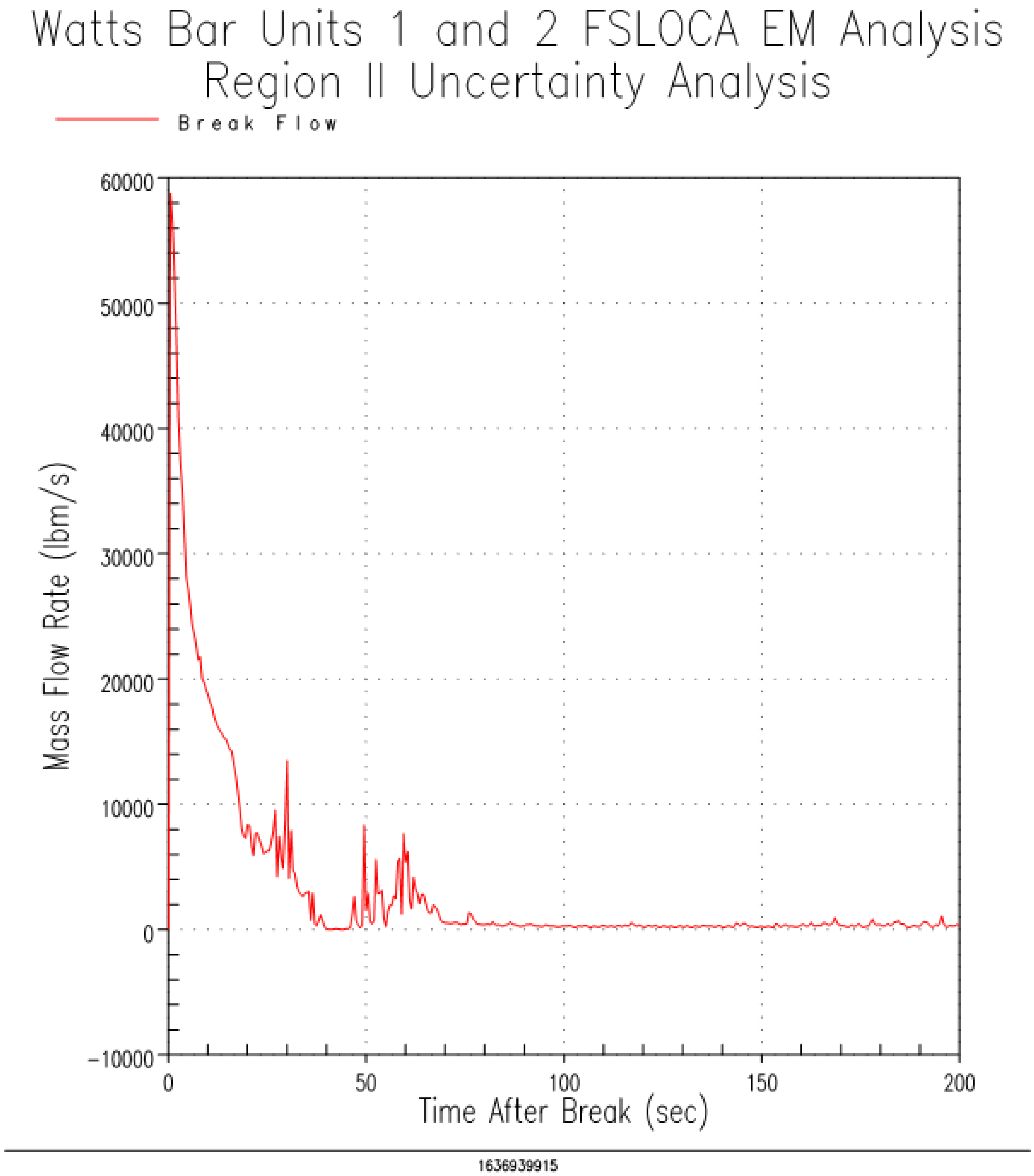

Figure 16: Watts Bar Units 1 and 2 Break Mass Flow Rate for Region II Analysis PCT Case

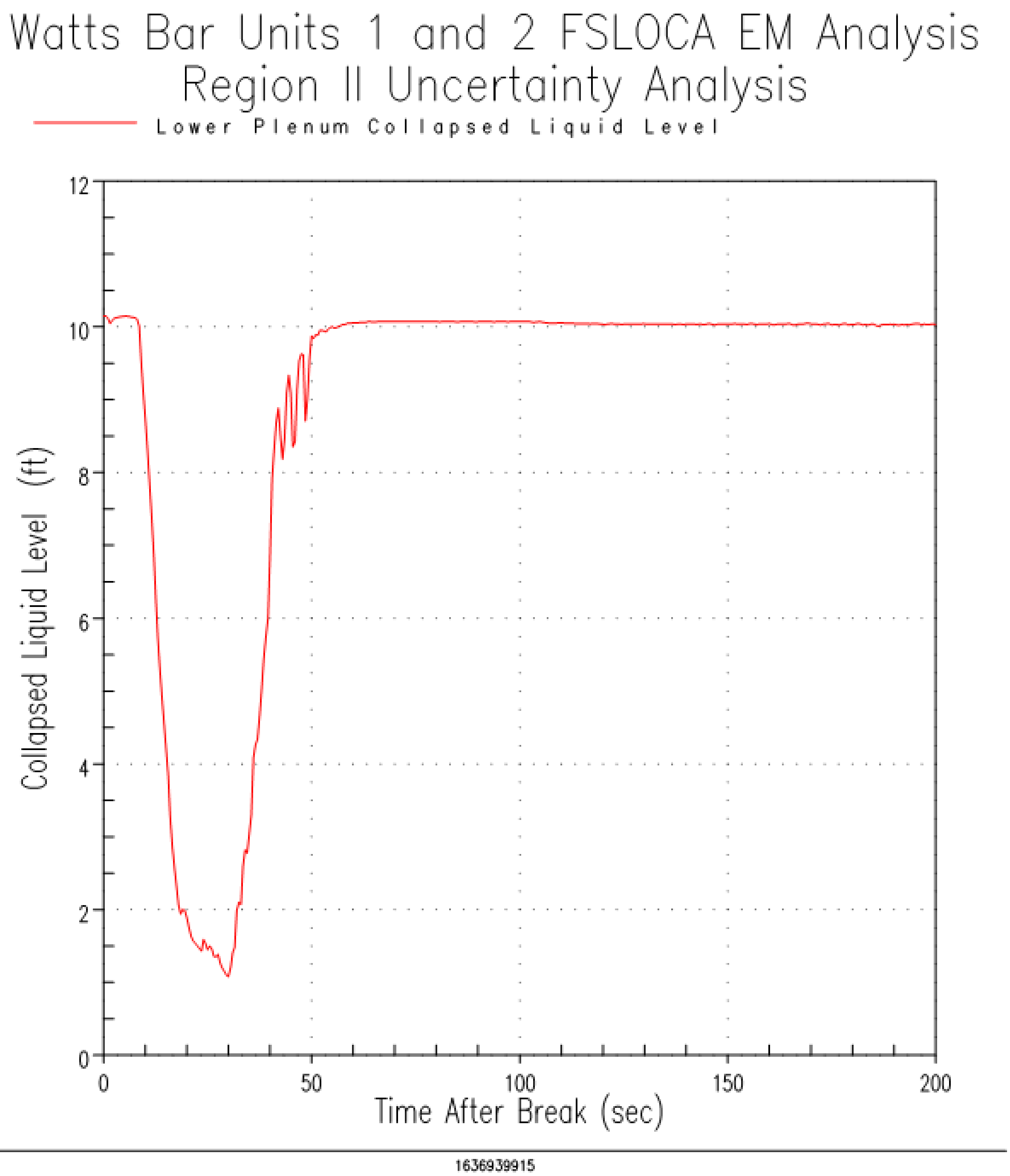

Figure 17: Watts Bar Units 1 and 2 Lower Plenum Collapsed Liquid Level (where 0 ft is the inside bottom of the reactor vessel) for Region II Analysis PCT Case

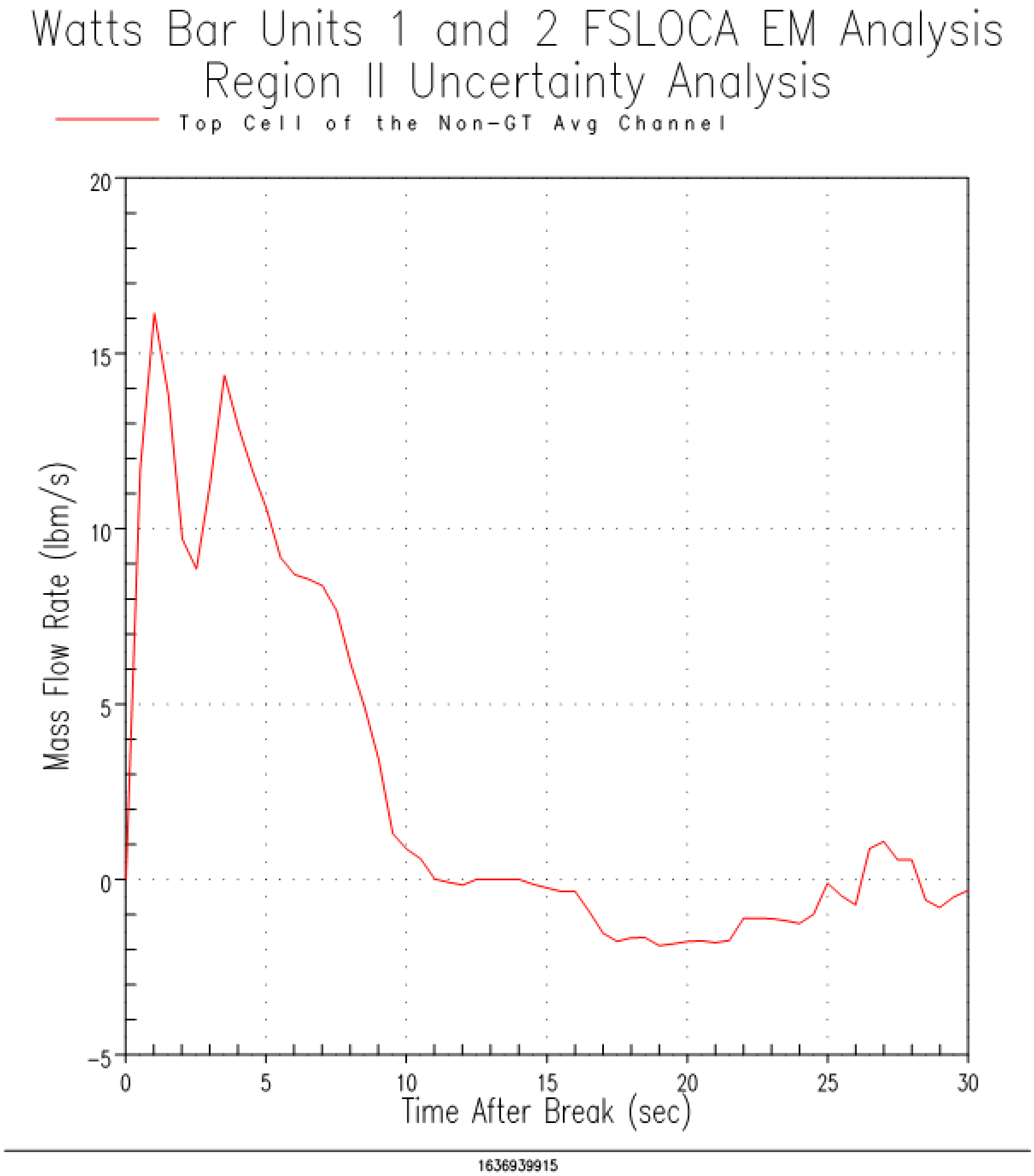

Figure 18: Watts Bar Units 1 and 2 Vapor Mass Flow Rate per Assembly at the Top Cell Face of the Core Average Channel Not Under Guide Tubes for Region II Analysis PCT Case

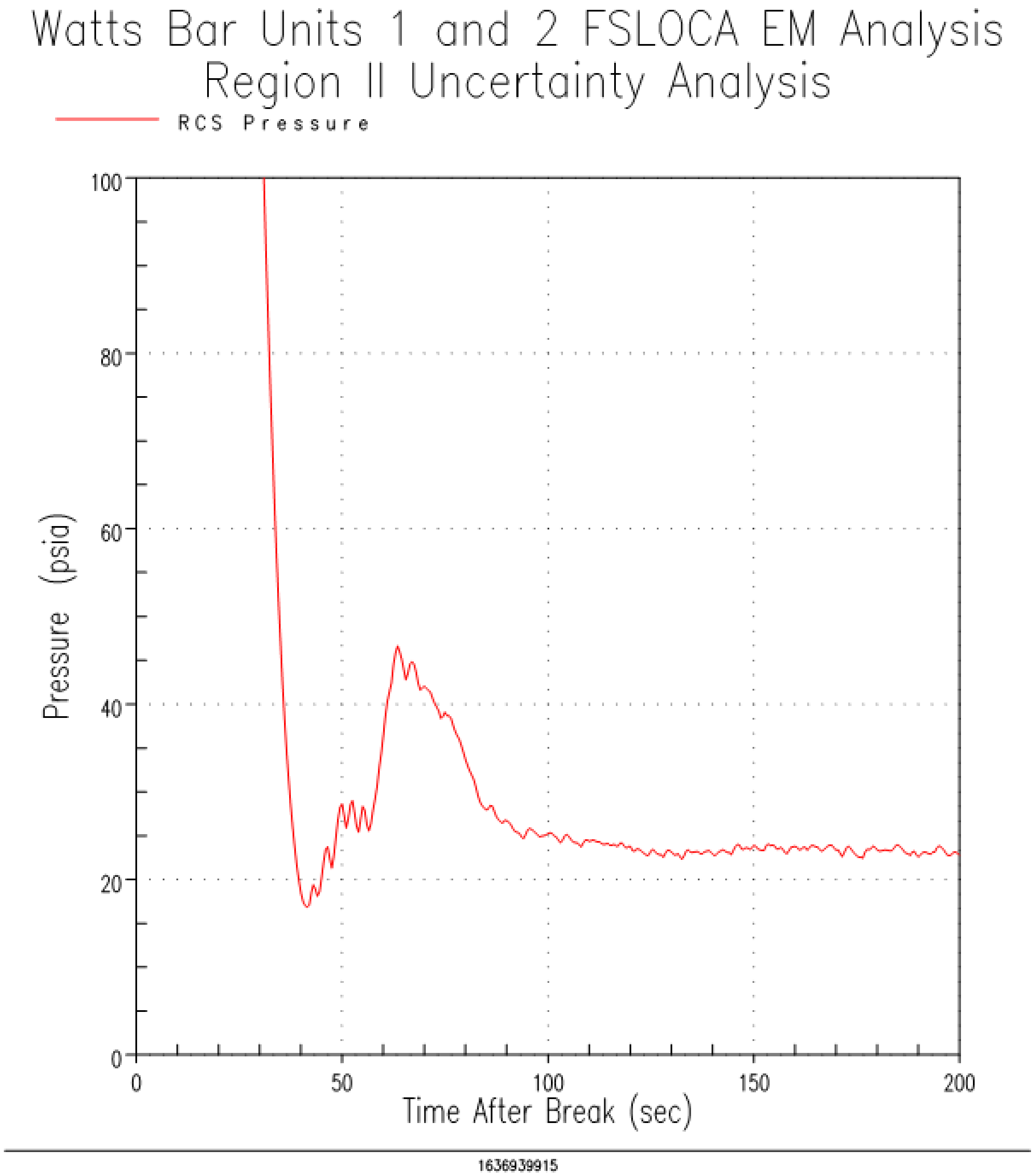

Figure 19: Watts Bar Units 1 and 2 RCS Pressure for Region II Analysis PCT Case

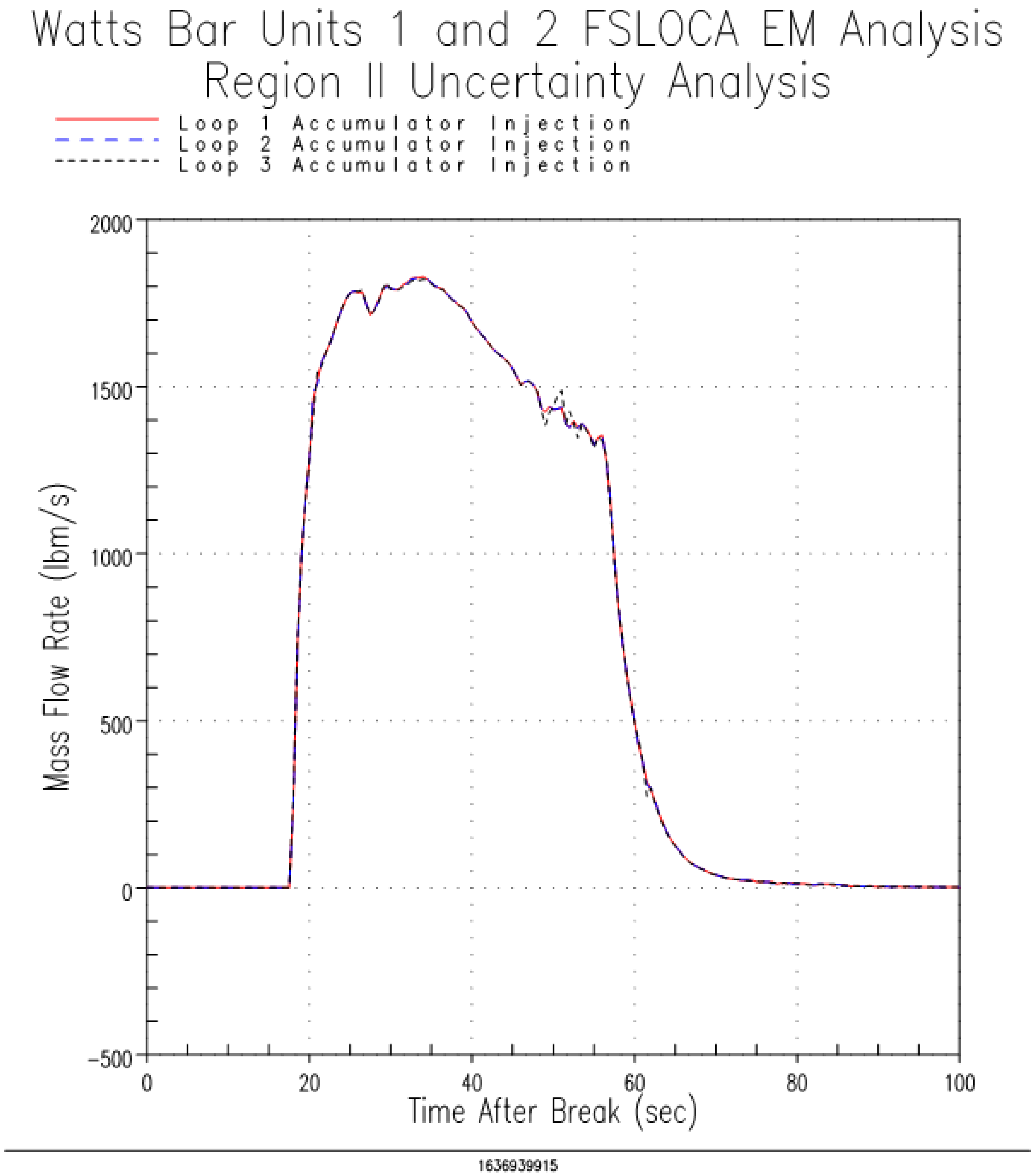

Figure 20: Watts Bar Units 1 and 2 Accumulator Injection Flow per Loop for Region II Analysis PCT Case

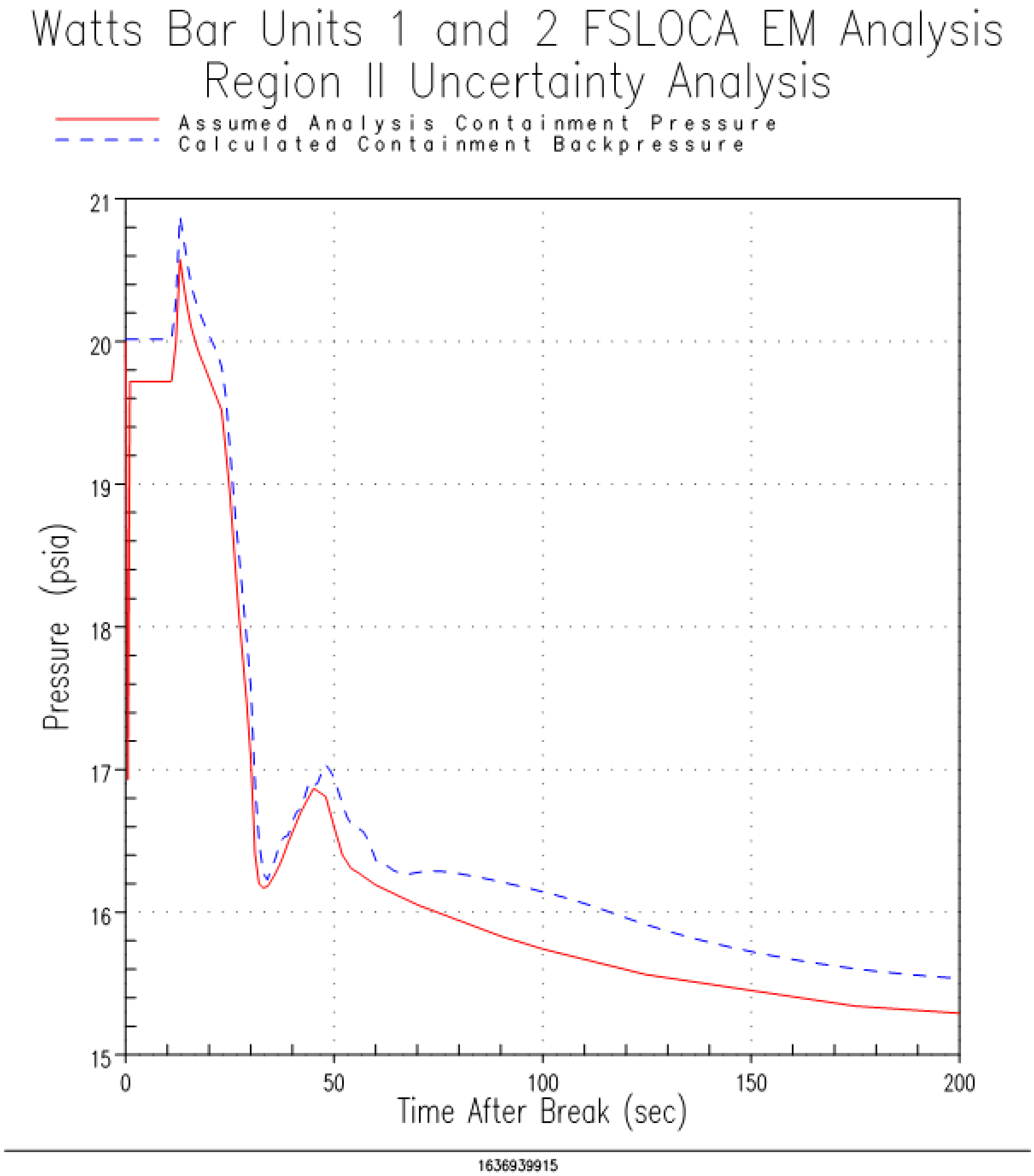

Figure 21: Watts Bar Units 1 and 2 Containment Pressure Comparison for Region II

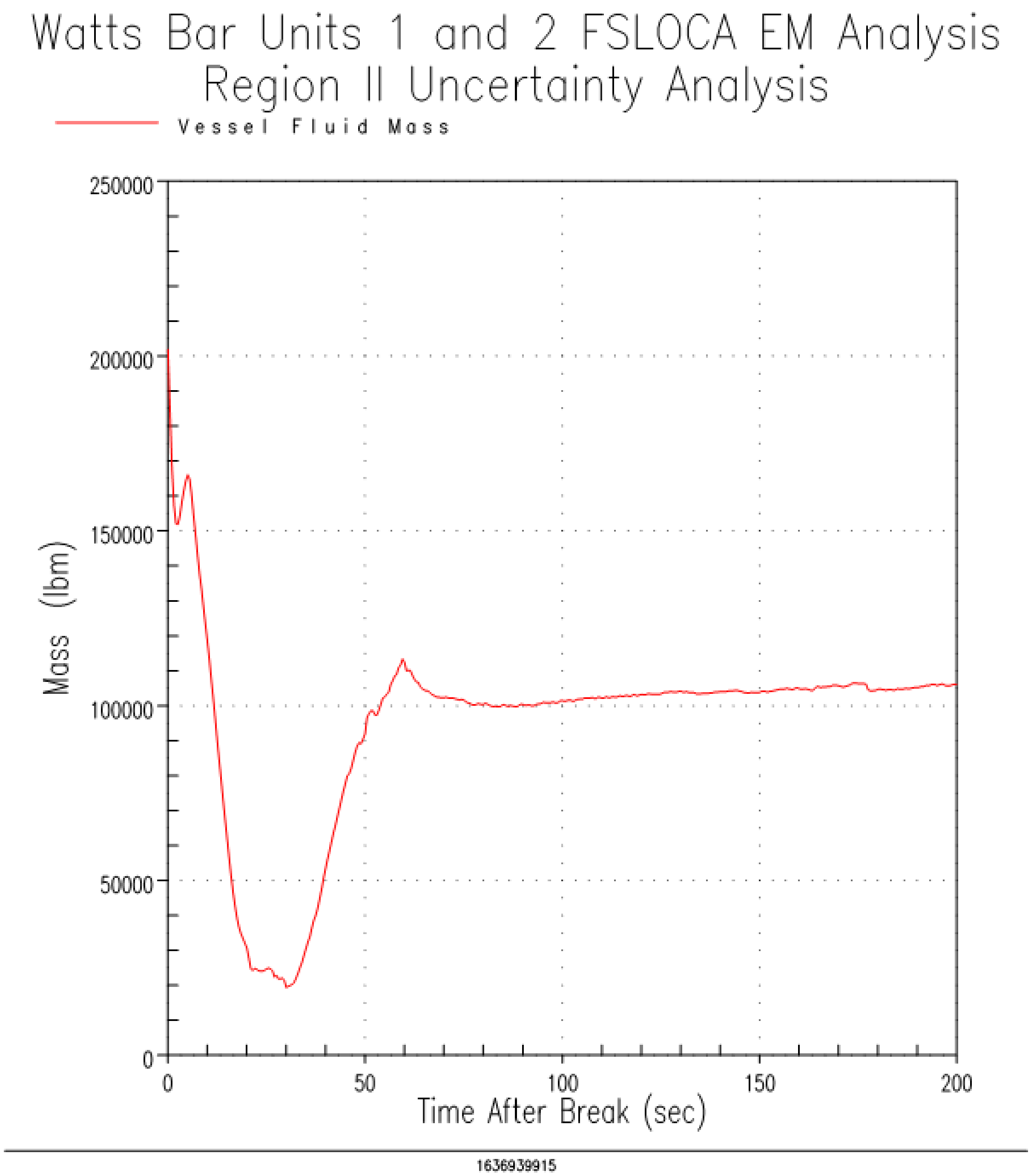

Figure 22: Watts Bar Units 1 and 2 Vessel Fluid Mass for Region II Analysis PCT Case

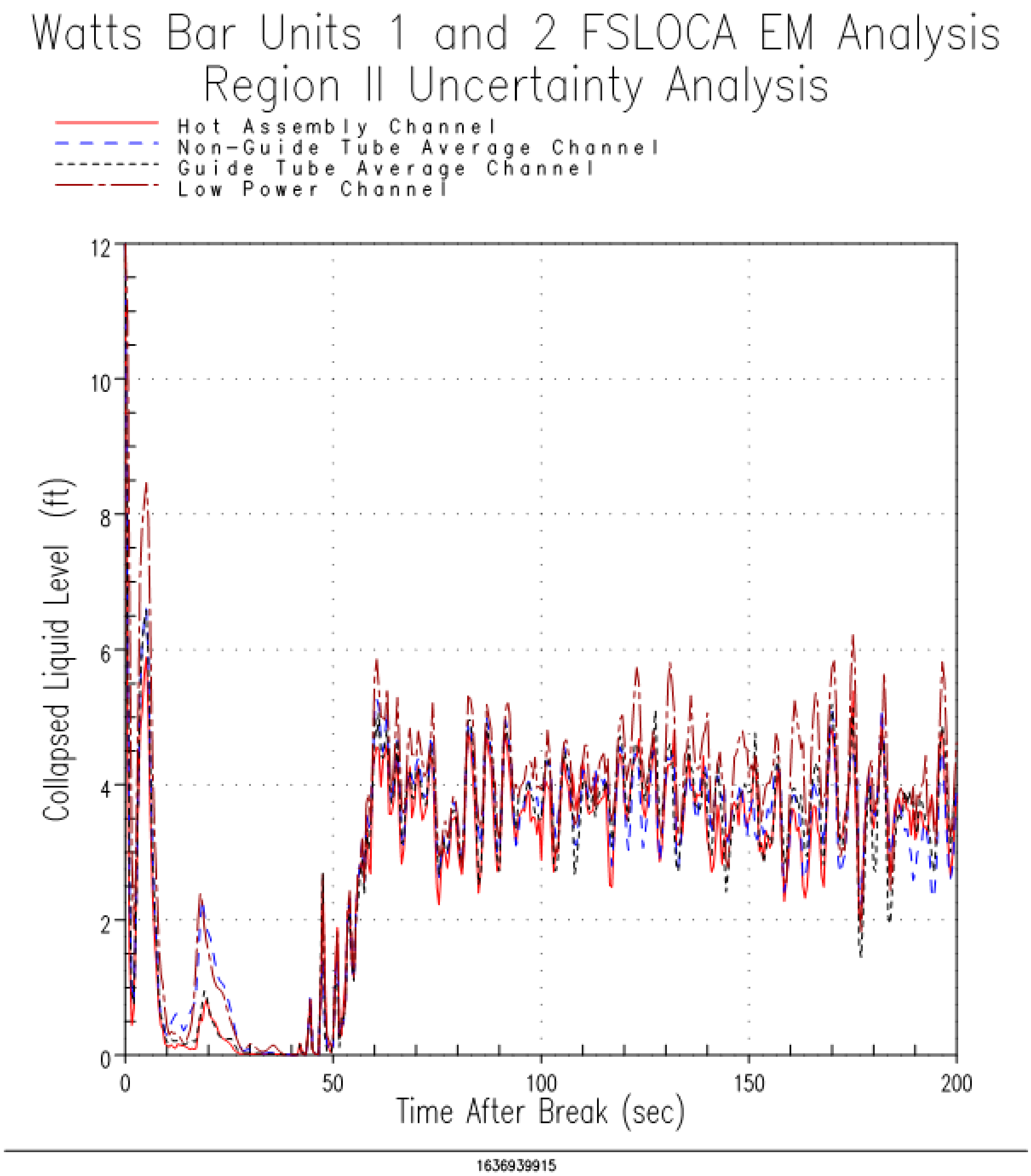

Figure 23: Watts Bar Units 1 and 2 Collapsed Liquid Levels (where 0 ft is bottom of active fuel, 12 ft is top of active fuel) for Each Core Channel for Region II Analysis PCT Case

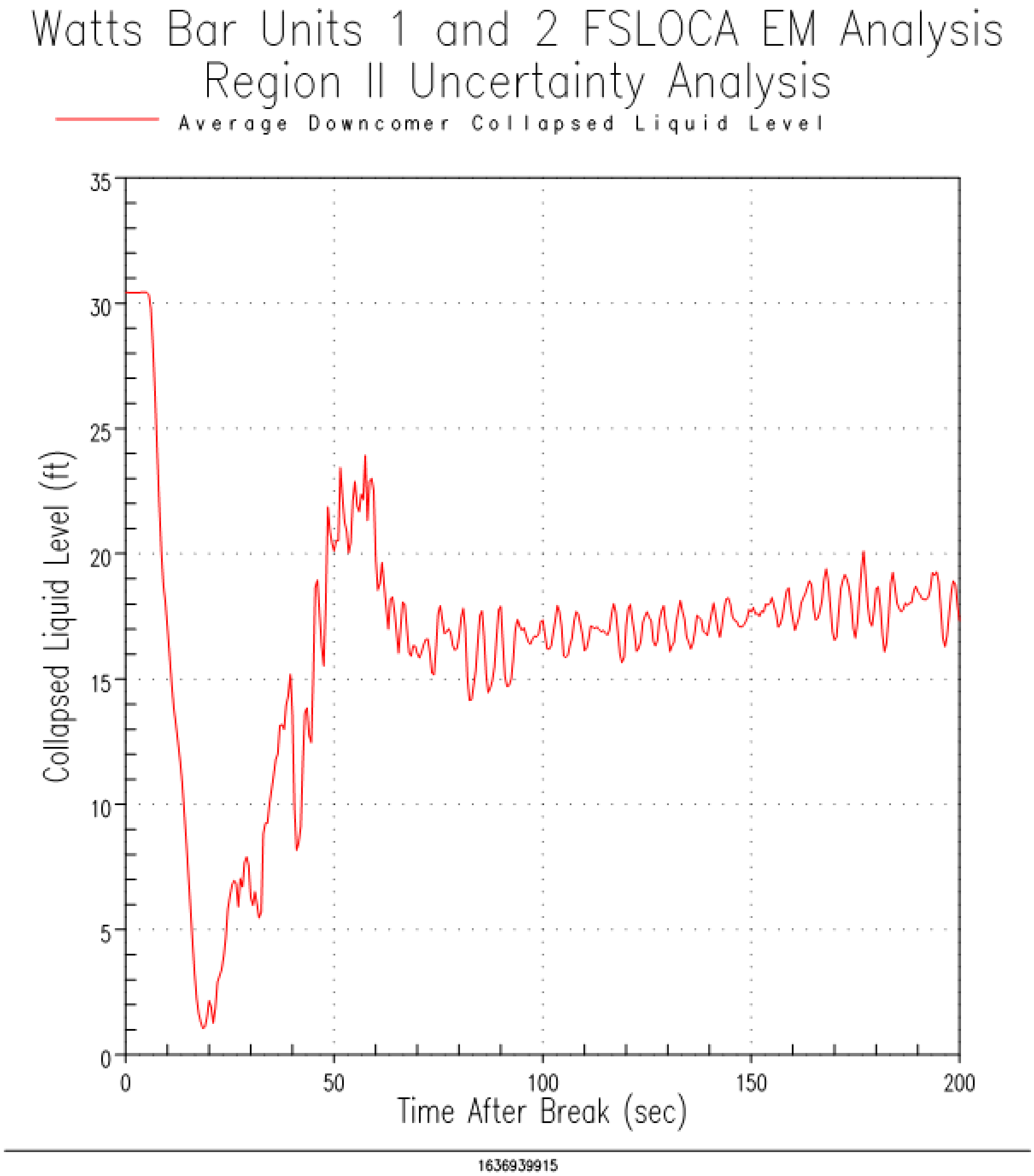

Figure 24: Watts Bar Units 1 and 2 Average Downcomer Collapsed Liquid Level (where 0 ft is the bottom of the upper tie plate) for Region II Analysis PCT Case

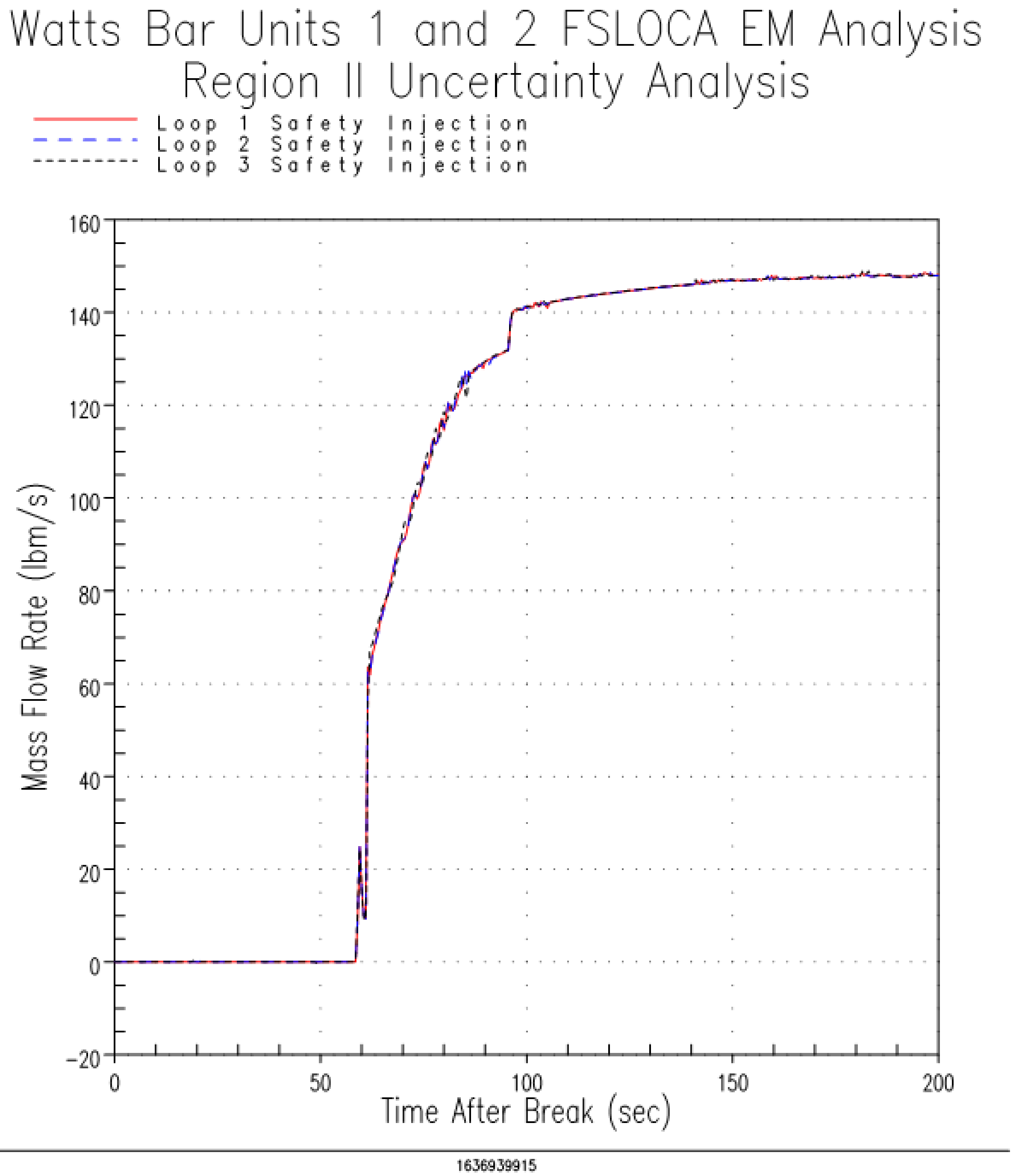

Figure 25: Watts Bar Units 1 and 2 Pumped SI Flow per Loop for Region II Analysis PCT Case

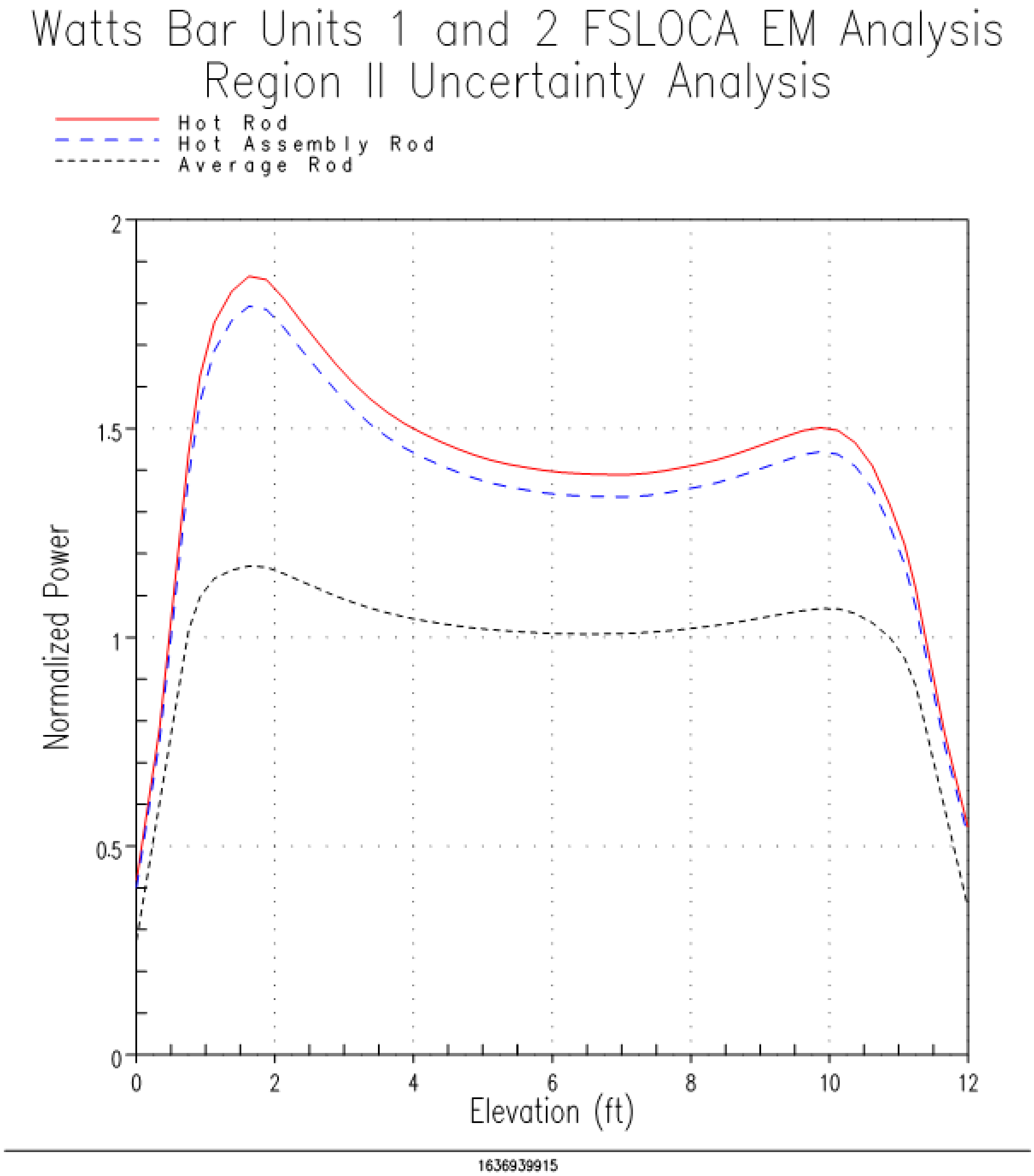

Figure 26: Watts Bar Units 1 and 2 Normalized Core Power Shapes for Region II Analysis PCT Case

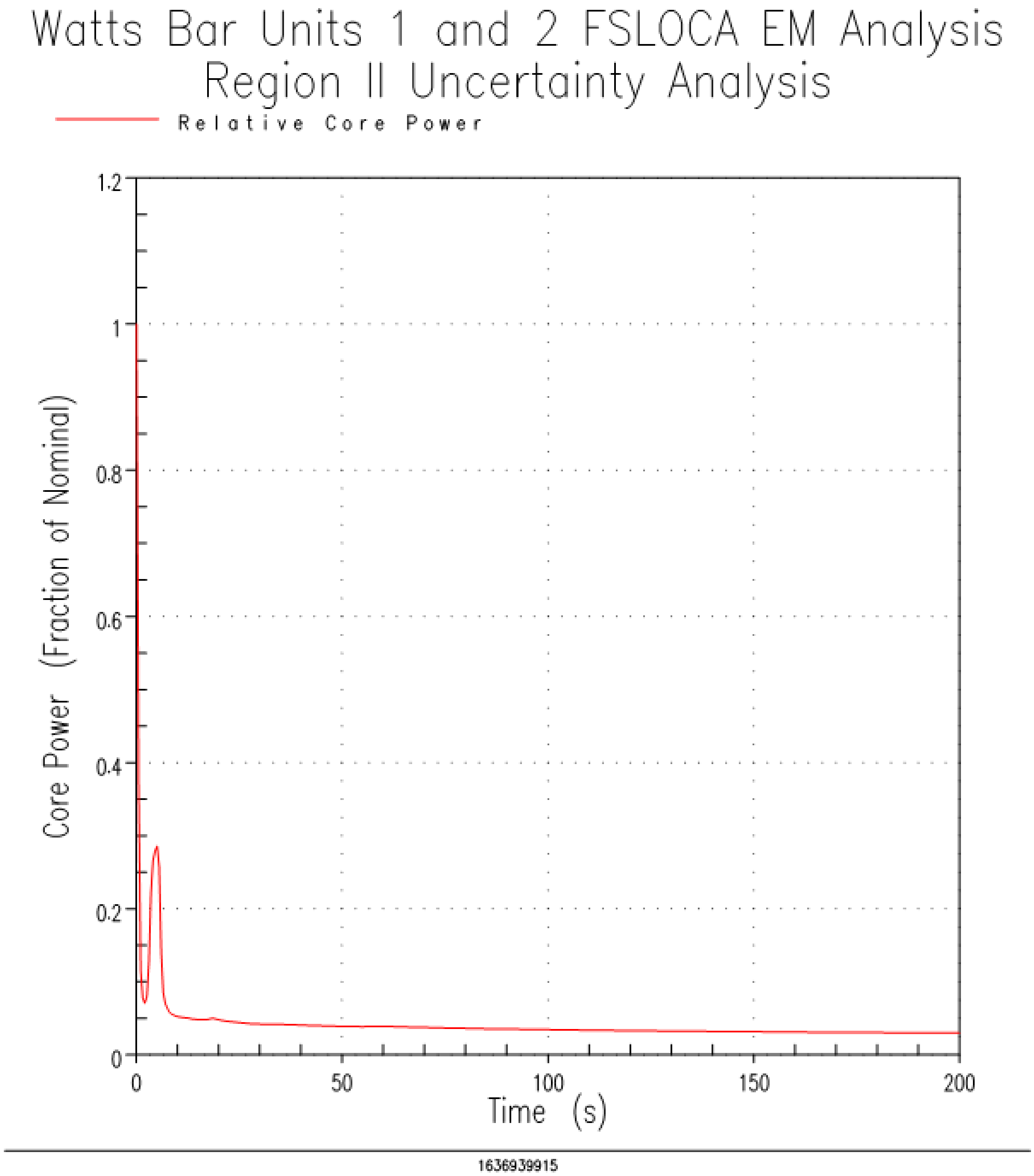

Figure 27: Watts Bar Units 1 and 2 Relative Core Power for Region II Analysis PCT Case

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNITS 1 AND 2

Westinghouse Electric Company LLC Application for Withholding Proprietary Information From Public Disclosure

Subject: Application to Implement the FULL SPECTRUM LOCA (FSLOCA) Methodology for Loss-of-Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04)

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Camille T. Zozula, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of CNL-21-010 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

(iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower-case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower-case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 07 Jan 2021

Camille Zozula

Camille T. Zozula, Manager
Regulatory Compliance & Corporate Licensing