~~Enclosure 1 is to be withheld from public disclosure under 10 CFR § 2.390. When separated from this enclosure, this letter is decontrolled.~~

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

August 6, 2013

10 CFR 50.46
10 CFR 2.390(b)(4)

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

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

Subject: WATTS BAR NUCLEAR PLANT UNIT 2 - FUEL THERMAL CONDUCTIVITY DEGRADATION, RESPONSE TO SUPPLEMENTAL SAFETY EVALUATION REPORT OPEN ITEM 61

References:

1. NRC Information Notice 2011-21, dated December 13, 2011, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation" (ADAMS Accession No. ML 113430785)

2. TVA Letter to NRC dated October 16, 2012, "Watts Bar Nuclear Plant (WBN) Unit 2 - 10 CFR 50.46 – Estimated Increase in Peak Clad Temperature (PCT) Due to the Effect of Fuel Pellet Thermal Conductivity Degradation and Peaking Factor Burndown Not Being Considered - 30 Day Report" (ADAMS Accession No. ML 12296A226)

3. Supplemental Safety Evaluation Report (SSER) 24, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2, Docket No. 50-391, Tennessee Valley Authority," published September 2011

The purpose of this letter is to provide information that addresses the Thermal Conductivity Degradation (TCD) effect on the Large-Break Loss of Coolant Accident (LOCA) analysis for Watts Bar Nuclear Plant (WBN) Unit 2. The information provided in Enclosures 1 and 4 completes TVA's response to Supplemental Safety Evaluation Report (SSER) Appendix HH, Open Item 61.

A002
NRR

The Nuclear Regulatory Commission (NRC) approved Westinghouse Best-Estimate Loss-of-Coolant Accident (BELOCA) Automated Statistical Treatment of Uncertainty Method (ASTRUM) methodology is based on the Westinghouse Improved Performance Analysis and Design (PAD) 4.0 fuel performance code. PAD 4.0 was licensed without explicitly considering fuel TCD with burnup.

Fuel performance data that accounts for fuel TCD was used as an input to the updated WBN Unit 2 LOCA analysis discussed herein. The new PAD fuel performance data was generated with an updated PAD model that includes explicit modeling of TCD. Therefore, the BELOCA analysis was updated to consider the fuel TCD effects cited in Reference 1. The discussion and results of this reanalysis are provided in Enclosure 1. Mark-ups of Final Safety Analysis Report (FSAR) Section 15.4.1 are provided in Enclosure 4. The FSAR changes will be incorporated in Amendment 110.

Enclosure 1 contains information proprietary to Westinghouse. Accordingly, TVA respectfully requests that this proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390. Enclosure 2 provides the supporting 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 Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390. NRC correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-11-3149 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Enclosure 3 provides the non-proprietary version of Enclosure 1. Enclosure 4 provides marked-up sections of the WBN Unit 2 FSAR incorporating the new information. Enclosure 5 provides the new commitment made in this letter.

If you have any questions, please contact me at (423) 365-1260 or Gordon Arent at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct to the best to my knowledge. Executed on the 6th day of August, 2013.

Respectfully,

R. A. Hruby, Jr.

Raymond A. Hruby, Jr.
General Manager, Technical Services
Watts Bar Unit 2

Enclosures:

1. Proprietary Attachment to Westinghouse Letter to TVA WBT-D-4396, dated July 23, 2013, "Thermal Conductivity Degradation (TCD) effect on the Large-Break LOCA Analysis for Watts Bar Nuclear Plant Unit 2"
2. Westinghouse Affidavit CAW-13-3755 for Withholding Proprietary Information from Public Disclosure
3. Non-Proprietary Attachment to Westinghouse Letter to TVA WBT-D-4396, dated July 23, 2013, "Thermal Conductivity Degradation (TCD) effect on the Large-Break LOCA Analysis for Watts Bar Nuclear Plant Unit 2"
4. WBN Unit 2 FSAR Chapter 15 Mark-up
5. New Commitment

cc (Enclosures):

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

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

ENCLOSURE 2

Tennessee Valley Authority
Watts Bar Nuclear Plant, Unit 2
Docket No. 50-391

WESTINGHOUSE AFFIDAVIT CAW-13-3755 FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Westinghouse

Westinghouse Electric Company
Engineering, Equipment and Major Projects
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: maurerbf@westinghouse.com
Proj letter: WBT-D-4396

CAW-13-3755

July 10, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WBT-D-4396 P-Attachment, "Thermal Conductivity Degradation Effect on the Large-Break LOCA Analysis for Watts Bar Unit 2" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-13-3755 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Tennessee Valley Authority.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference CAW-13-3755, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

Bradley F. Maurer, Principal Engineer
Plant Licensing

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared Bradley F. Maurer, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

BF Maurer

Bradley F. Maurer, Principal Engineer
Plant Licensing

Sworn to and subscribed before me
this 10th day of July 2013

Renee' Giampole

Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Renee Giampole, Notary Public
Penn Township, Westmoreland County
My Commission Expires September 25, 2013

(1) I am Principal Engineer, Plant Licensing, in Engineering, Equipment and Major Projects, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(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 the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, 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.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

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.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WBT-D-4396 P-Attachment, "Thermal Conductivity Degradation Effect on the Large-Break LOCA Analysis for Watts Bar Unit 2" (Proprietary), for submittal to the Commission, being transmitted by Tennessee Valley Authority letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for Tennessee Valley Authority use for Watts Bar Unit 2 to close out certain NRC open items related to the Large Break Loss-of-Coolant Accident analysis, and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Provide input to the U.S. Nuclear Regulatory Commission for review of Watts Bar Unit 2 (WBT) Completion Project submittals.

(b) Provide information on fuel thermal conductivity degradation results for the LB BELOCA.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of the information to its customers for the purpose of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the technology to its customer in its licensing process.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

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 calculations 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.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests to close out certain NRC open items related to the Large Break Loss-of-Coolant Accident analysis.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary 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 (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

Tennessee Valley Authority

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC Document Control Desk:

Enclosed are:

1. One (1) copy of WBT-D-4396 P-Attachment, Revision 0, "Thermal Conductivity Degradation Effect on the Large-Break LOCA Analysis for Watts Bar Unit 2" (Proprietary)
2. One (1) copy of WBT-D-4396 NP-Attachment, Revision 0, "Thermal Conductivity Degradation Effect on the Large-Break LOCA Analysis for Watts Bar Unit 2" (Non-Proprietary)

Also enclosed is the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-13-3755, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC, 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 Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested 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 affidavit should reference CAW-13-3755 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

ENCLOSURE 3
Tennessee Valley Authority
Watts Bar Nuclear Plant, Unit 2
Docket No. 50-391

NON-PROPRIETARY ATTACHMENT TO WESTINGHOUSE LETTER TO TVA WBT-D-4396, DATED JULY 23, 2013, "THERMAL CONDUCTIVITY DEGRADATION (TCD) EFFECT ON THE LARGE-BREAK LOCA ANALYSIS FOR WATTS BAR NUCLEAR PLANT UNIT 2"

NP-Attachment to WBT-D-4396
Thermal Conductivity Degradation Effect on the Large-Break LOCA Analysis for Watts Bar Unit 2

Thermal Conductivity Degradation Effect on the Large-Break LOCA Analysis for Watts Bar Unit 2

1.0 Introduction:

This report is supplemental to FSAR Section 15.4.1.1 [1] and serves to communicate the results of an updated LBLOCA analysis performed to address the fuel Thermal Conductivity Degradation Industry Issue [4].

The Nuclear Regulatory Commission (NRC) approved Westinghouse Best-Estimate Loss-of-Coolant Accident (BELOCA) Automated Statistical Treatment of Uncertainty Method (ASTRUM) methodology [2] is based on the PAD 4.0 fuel performance code [3]. PAD 4.0 was licensed without explicitly considering fuel thermal conductivity degradation (TCD) with burnup. Explicit modeling of TCD in the fuel performance code leads directly to increased fuel temperatures (pellet radial average temperature) as well as other fuel performance related effects beyond beginning-of-life. Since PAD provides input to the large-break LOCA analysis, this will tend to increase the stored energy at the beginning of the simulated large-break LOCA event. This in turn leads to an increase in Peak Cladding Temperature (PCT) if there is no provision to credit off-setting effects.

The Watts Bar Unit 2 PCT was previously ([1], Table 15.4-18b) calculated to be 1552°F, the Maximum Local Oxidation (MLO) was calculated to be 1.04% and the Core-Wide Oxidation (CWO) was calculated to be 0.0% with the currently licensed Westinghouse BELOCA methodology [2]. Thus, the analysis demonstrated margin to the analysis limits of:

PCT	2200°F
MLO	17%
CWO	1%

Fuel performance data that accounts for fuel TCD was used as input to the updated Watts Bar Unit 2 BELOCA analysis discussed herein. The new PAD fuel performance data was generated with an updated PAD model that includes explicit modeling of TCD. Therefore, the BELOCA analysis was updated to consider the fuel TCD effects cited in NRC Information Notice 2011-21 [4].

2.0 Input Parameters, Assumptions and Acceptance Criteria:

No updates to design inputs and plant operating ranges to gain large-break LOCA margin were utilized in this updated analysis to show compliance with the 10 CFR 50.46 acceptance criteria while maintaining a margin of safety to the prescribed limits. The acceptance criteria and results of the updated BELOCA analysis considering TCD effects are discussed in Section 4.0. The base input assumptions are provided in Tables 15.4-14,15,16,19,23 and Figure 15.4-56 of the FSAR [1], with the limited exception of core peaking factors as noted below.

In order to mitigate the impact of the increasing effect of pellet TCD with burnup, the large-break LOCA evaluation of 2nd Cycle fuel utilized reduced peaking factors from those shown in FSAR Table 15.4-19 [1]. The reduced peaking factors are limited to the following application:

> Burndown credit for the hot rod and hot assembly is taken for higher burnup fuel in the 2nd and 3rd cycle of operation. The Watts Bar Unit 2 peaking factor values utilized in this updated analysis are shown in Table 1. Note that the beginning to middle of life values are retained at their original [1] values.

3.0 Description of Analysis and Evaluations:

The purpose of this update to the analysis is to consider fuel performance inputs that explicitly model TCD to show compliance with the 10 CFR 50.46 acceptance criteria while maintaining a margin of safety to the prescribed limits. The updated BELOCA analysis considering the effects of TCD is supplemental to information provided in FSAR Section 15.4.1.1 [1].

[

]^{a,c}

The updated analysis also credited peaking factor burndown to evaluate higher burnup fuel in its second/third cycle of irradiation. Evaluation of fuel in its second/third cycle of irradiation is beyond the first cycle considered in the approved ASTRUM Evaluation Model (EM), but was considered in the updated analysis when explicitly modeling TCD to demonstrate that conformance to the acceptance criteria is met for the second/third cycle fuel. Physically, accounting for TCD leads to an increase in fuel temperature as the fuel is burned, while accounting for peaking factor burndown leads to a reduction in fuel temperature as the fuel is burned. The compensating nature of these phenomena is considered in the updated analysis in order to appropriately capture the effect of TCD in the updated Watts Bar Unit 2 BELOCA analysis.

The analysis was updated by re-running all 124 cases from the original ASTRUM analysis (FSAR Section 15.4.1.1 [1]) in both the first cycle and the second cycle. Therefore the same non-parametric order statistics singular statement of a 95th percentile at the 95-percent confidence joint probability for PCT, MLO and CWO of an ASTRUM re-analysis is ensured for the Watts Bar Unit 2 updated analysis. In addition, the uncertainty parameters seed treatment is [

]^{a,c}

The confirmatory configuration and the conservatively low containment backpressure from the original ASTRUM run set (FSAR Section 15.4.1.1 [1]) were also re-evaluated considering the effects of TCD. The limiting plant configuration was determined to remain the same. Details of the FSAR analysis confirmatory suite limiting plant configuration study were not previously presented in the FSAR Section 15.4.1.1, but are included herein as Table 5 to support this conclusion. First, it is reinforced that no miscellaneous plant configuration changes have been introduced, meaning the confirmation study evaluation only needs to focus to physical TCD effects.

[

]^{a,c}

The conservatively low containment backpressure from the original analysis remains bounding since the core stored energy increases when explicitly modeling fuel TCD, which would tend to increase energy released through the break and hence increase the containment pressure.

IFBA fuel

[

]^{a,c}

4.0 Results:

The Watts Bar Unit 2 PCT-limiting transient is a double-ended cold leg guillotine break when considering fuel TCD and the peaking factor burndown provided in Table 1. Table 2 summarizes the results of the updated BELOCA analysis considering the effects of TCD. Table 3 provides the sequence of events for the PCT-limiting transient from the updated analysis. Further IFBA and non-IFBA PCT results for both cycles are given in Table 4.

Figures 1-15 provide a variety of transient responses for the limiting HOTSPOT PCT transient (2nd Cycle, run073, non-IFBA) and are generally self explanatory in nature. Figures 2-15 are from the associated WCOBRA/TRAC response. Figure 1 multiplots the following:

HOTSPOT Clad Temperature at the limiting PCT elevation (observe the peak at 1766°F)
WCOBRA/TRAC PCT. This PCT represents the highest clad temperature at any elevation.

These Figures are equivalent to FSAR Figures 15.4-41 through 15.4-55 [1].

FSAR Figures 15.4-40b,c,d,e,f,g [1] remain equally appropriate for the TCD analysis because the containment backpressure study did not need repeated as discussed in Section 3.0.

The updated TCD analysis compares to the prior Analysis as follows:

Case	PCT (°F)	
TCD 1st Cycle Runset, Limiting run100	1580	(non-IFBA)
TCD 2nd Cycle Runset, Limiting run073	1766	(non-IFBA)

TCD Limiting IFBA Run, 2nd Cycle run073	1763	(IFBA)
FSAR [1] Limiting run018	1552	(non-IFBA)

10 CFR 50.46 Requirements

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 1766°F, the updated analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F," is demonstrated. The result is shown in Table 2.

(b)(2) The maximum local cladding oxidation corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting transient MLO for the limiting case is 1.99%, the updated 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 result is shown in Table 2.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total power census includes many lower power assemblies. The CWO value can be conservatively chosen as that calculated for the limiting HAR, **0.08%**. A detailed CWO calculation is not needed due to the margin in the conservatively obtained result. Therefore, the updated analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent," is demonstrated. The result is shown in Table 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 amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The approved methodology [5] specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 44 assemblies on the core periphery. This situation is not calculated to occur for Watts Bar Unit 2 per FSAR 15.4.1.1.6 [1] prior to TCD considerations, and this conclusion is not affected by the modeling of fuel TCD. Therefore, acceptance criterion (b)(4) remains satisfied.

(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 continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at these plants to maintain long-term cooling remain unchanged due to the modeling of fuel TCD, as follows:

The primary impact of TCD in the fuel potentially important to Long Term Cooling (LTC) is an increase in initial fuel pellet temperature. This in turn leads to a higher amount of stored energy at the initiation of the LOCA event. Initial stored energy is not important to LTC evaluations as these evaluations only consider decay heat removal during the sump recirculation phase of emergency core cooling system (ECCS) operation. The increased stored energy in the fuel due to higher fuel pellet temperature is a short term effect that does not persist into the LTC phase of ECCS performance evaluations; therefore, the heat source remains limited to decay heat for LTC evaluations. Consequential impacts of higher fuel pellet temperature such as higher fuel rod internal pressure also have no impact on LTC evaluations as fuel cladding temperatures are maintained well below the threshold for cladding rupture such that cladding burst and blockage does not occur during LTC. Based on the above, it is shown that no additional LTC analysis is required to assess TCD for Watts Bar Unit 2.

Results Summary

An update to the analysis was performed considering fuel performance inputs that explicitly model TCD and the inherently associated peaking factors burndown credit for 2nd cycle analysis only (Table 1), including [ ]a,c IFBA fuel product to show compliance with the current 10 CFR 50.46 acceptance criteria while maintaining a margin of safety to the prescribed limits. For both the 1st cycle study and the 2nd cycle study, the ASTRUM EM [2] required 124 runs were executed using the same random seed. In this way, the integrity of the TCD analysis is maintained and the difference between the new 95/95 estimate herein and the previous pre-TCD estimate FSAR [1] Section 15.4.1.1 is the singular effect of TCD. Based on the results from the updated BELOCA analysis (see Table 2), it is concluded that Watts Bar Unit 2 continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

Additional Information:

Additional Information on treatment of burnup and fuel in its second and third cycle of irradiation

The NRC-approved Westinghouse ASTRUM Methodology [2] assumes a LOCA to be [

]a,c Please refer to Section 11-2-2 of the ASTRUM Topical [2] for more information. Although a small amount of peaking factor burndown credit might have been available for some of the higher burnup cases of the 1st cycle runset (Table 1), no credit was taken.

[

]a,c Since the analyzed burnup range covered up to the [ ]a,c design limit, fuel in its first, second or third cycles of operation are covered.

	Hot Rod Burnup (MWD/MTU) Minimum	Maximum
1st Cycle Study	[	]a,c
2nd Cycle Study	[	]a,c

	Hot Rod Burnup (MWD/MTU)
Overall Limiting PCT Case 073 (2nd Cycle)	[ ]a,c
Case 073 1st Cycle	[ ]a,c

References:

1. WBT-D-1373. WEC to TVA ASTRUM FSAR Package.
   (Recommend TVA convert this to the counterpart submittal to the NRC.)

2. Westinghouse Report WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," January 2005. (Westinghouse Proprietary Class 2)

3. Westinghouse Report WCAP-15063-P-A, Revision 1 with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July 2000. (Westinghouse Proprietary Class 2)

4. NRC Information Notice 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," December 13, 2011. (NRC ADAMS Accession Number ML113430785)

5. Westinghouse Report WCAP-12945-P-A, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998. (Westinghouse Proprietary Class 2)

Table 1	**Watts Bar Unit 2 Best-Estimate Large-Break LOCA Updated Analysis Considering the Effects of Thermal Conductivity Degradation - Summary of Peaking Factor Burndown Utilized**		
Hot Rod Burnup (MWD/MTU)	**FdH (1) (with uncertainties)**	**FQ Transient (with uncertainties)**	**FQ Steady-state (without uncertainties)**
0	1.65 (2)	2.50 (2)	2.00 (2)
30000	1.65 (2)	2.50 (2)	2.00 (2)
60000	1.525	2.25	1.800
62000	1.525	2.25	1.800
Note 1 The standard BELOCA assumption of [ ]^{a,c}			
Note 2: Same Value as FSAR Table 15.4-19 (Note FQ SS is titled 'SS depletion' therein)			

Table 2	**Watts Bar Unit 2 Best-Estimate Large-Break LOCA Updated Analysis Considering the Effects of Thermal Conductivity Degradation – Comparison of Results to Current 10 CFR 50.46(b) Acceptance Criteria**	
	Result	**Acceptance Criterion**
95/95 PCT[1]	1766	< 2200°F
95/95 Transient MLO[2]	1.99	< 17%
95/95 CWO[3]	0.08	< 1%
Coolable Geometry	Criterion Met (See Section 4.0 (b)(4) herein)	Remains Coolable
Long-Term Cooling	See Long Term Cooling TCD assessment in Section 4.0 (b)(5) herein	

Notes:

1. Peak Cladding Temperature
2. Maximum Local Oxidation, transient
3. Core-wide Oxidation

Table 3	Watts Bar Unit 2 Best-Estimate Large-Break LOCA Updated Analysis Considering the Effects of Thermal Conductivity Degradation - Sequence of Events for the Limiting PCT Transient
Event	**Time After Break (sec)**
Start of Transient	0
Safety Injection Signal	5
Accumulator Injection Begins	10
End of Blowdown	11
Bottom of Core Recovery	36
Accumulator Empty[1]	43
Safety Injection Begins	60
PCT Occurs	190
End of Analysis Time	400

Notes:

1. Accumulator switches from liquid to injection
2. Limiting Case is 073, Non-IFBA, of 2^{nd} Cycle study

Table 4	Watts Bar Unit 2 Best-Estimate Large-Break LOCA Updated Analysis Considering the Effects of Thermal Conductivity Degradation - Detailed Results					
MLO						
Rank	**Run #**	**Non-IFBA MLO (%)**	**Non-IFBA Hot Rod Burnup (MWD/MTU)**	**Run #**	**IFBA MLO (%)**	**IFBA Hot Rod Burnup (MWD/MTU)**
1st Cycle						
1	073	1.83	14785	073	1.76	14785
2nd Cycle						
1	104	1.45	52673	104	**1.99**	52673
2	73	1.37	46909	93	1.48	61773
3	18	1.06	45156	73	1.34	46909
4	100	0.88	46759	18	1.08	45156
5	93	0.81	61773	118	0.97	37365
PCT						
Rank	**Run #**	**Non-IFBA HOTSPOT PCT (°F)**	**Non-IFBA Hot Rod Burnup (MWD/MTU)**	**Run #**	**IFBA HOTSPOT PCT (°F)**	**IFBA Hot Rod Burnup (MWD/MTU)**
1st Cycle						
1	100	1580	14940	100	1579	14940
2nd Cycle						
1	73	**1766**	46909	73	1763	46909
2	100	1646	46759	104	1646	52673
3	104	1630	52673	100	1645	46759
4	42	1609	57001	118	1637	37365
5	118	1596	37365	42	1607	57001

Table 5	Watts Bar Unit 2 Best-Estimate Large-Break LOCA Updated Analysis Considering the Effects of Thermal Conductivity Degradation - Original FSAR [1] WCOBRA/TRAC Confirmatory Studies Results				
	Confirmatory Study Parameters			**WC/T Results**	
Confirmatory Run #	**SGTP (%)**	**PLOW**	**LOOP?**	**Hot Rod PCT (°F)**	**Hot Rod PCT Time (sec after break)**
002	High (10)	High (0.8)	LOOP	1229	3.1
004			No-LOOP	1200	2.9
006[1]		Low (0.2)	**LOOP**	**1298**	**265**
008			No-LOOP	1210	248
010	Low (0)	High (0.8)	LOOP	1221	3.0
012			No-LOOP	1196	2.9
014		Low (0.2)	LOOP	1209	2.9
016			No-LOOP	1183	2.7

Note:

Watts Bar does not have an RCS T_{avg} window.

1. Denotes WCOBRA/TRAC limiting PCT/Reference Transient

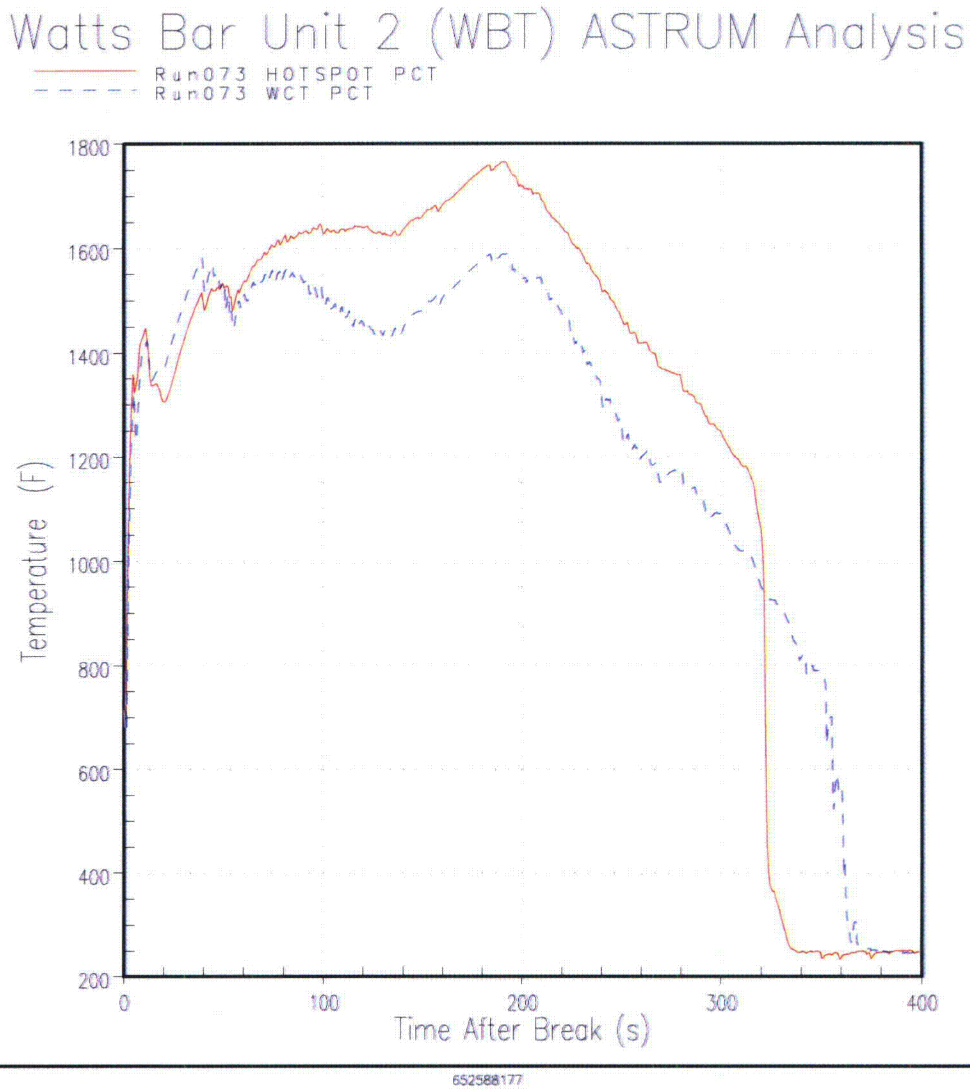

Figure 1
Watts Bar Unit 2 Limiting PCT Case (2nd Cycle, Run073, non-IFBA) HOTSPOT Clad Temperature at the Limiting Elevation and WC/T PCT

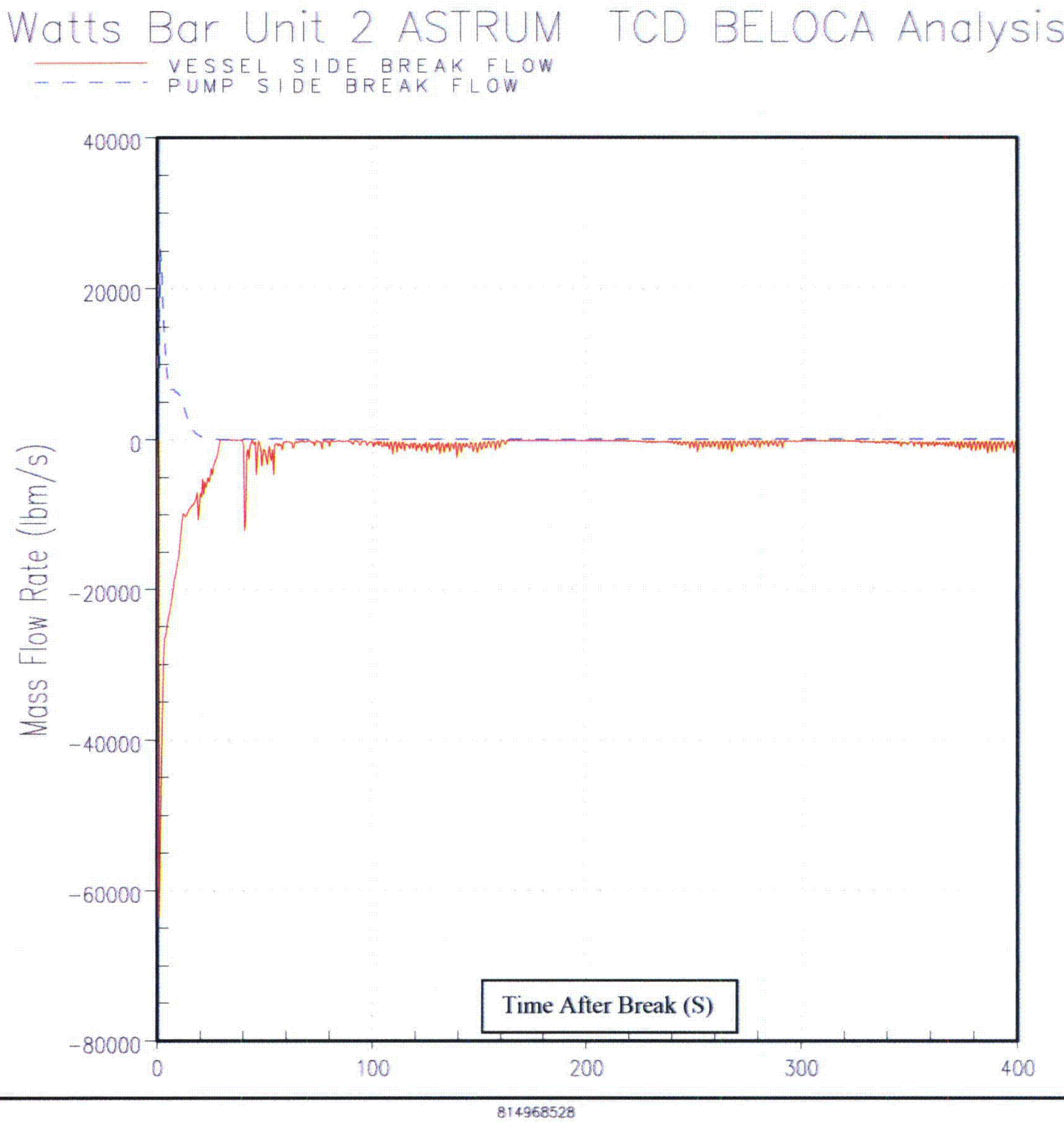

Figure 2
Watts Bar Unit 2 Limiting PCT Case Break Flow

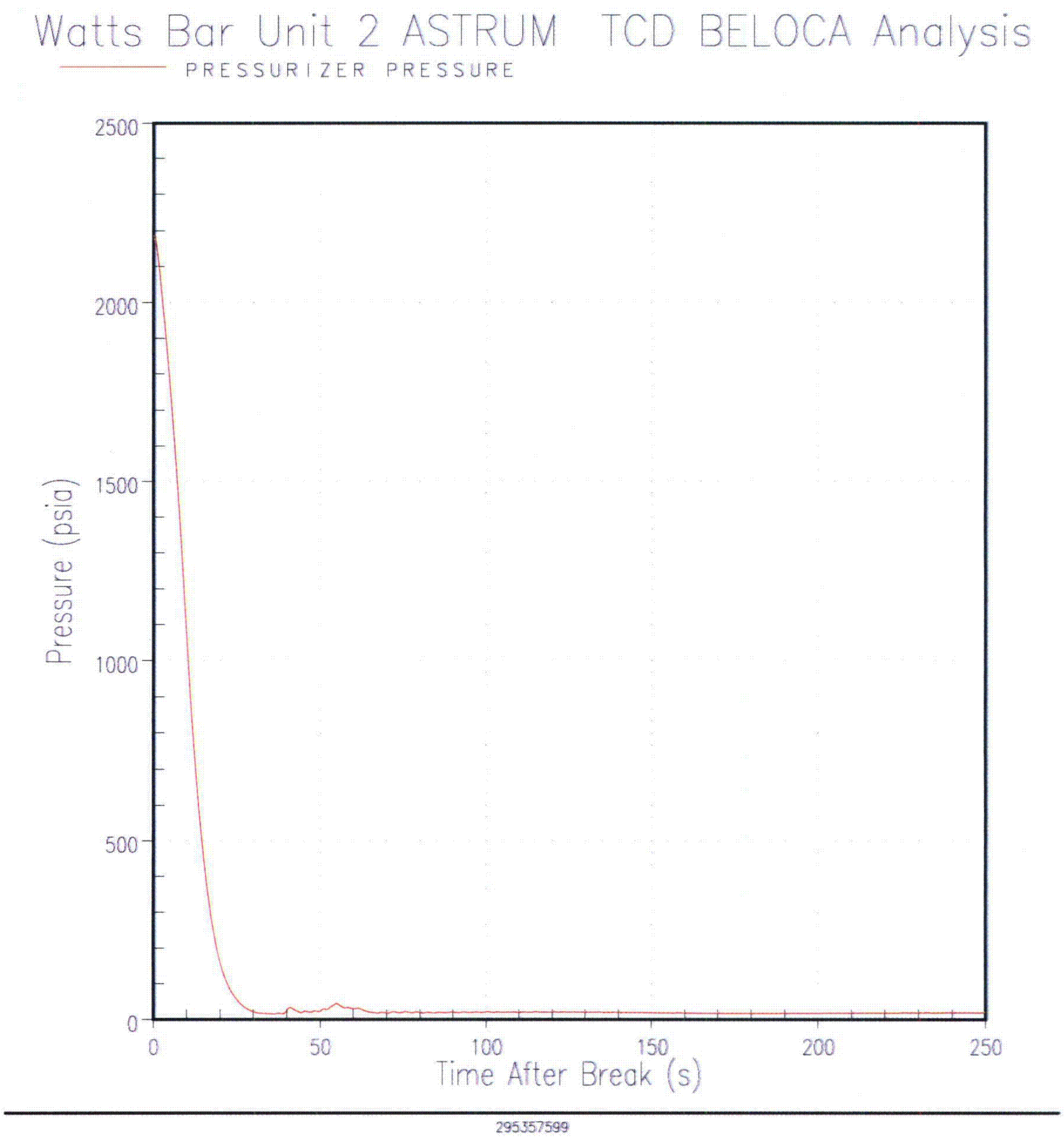

Figure 3
Watts Bar Unit 2 Limiting PCT Case Pressurizer Pressure

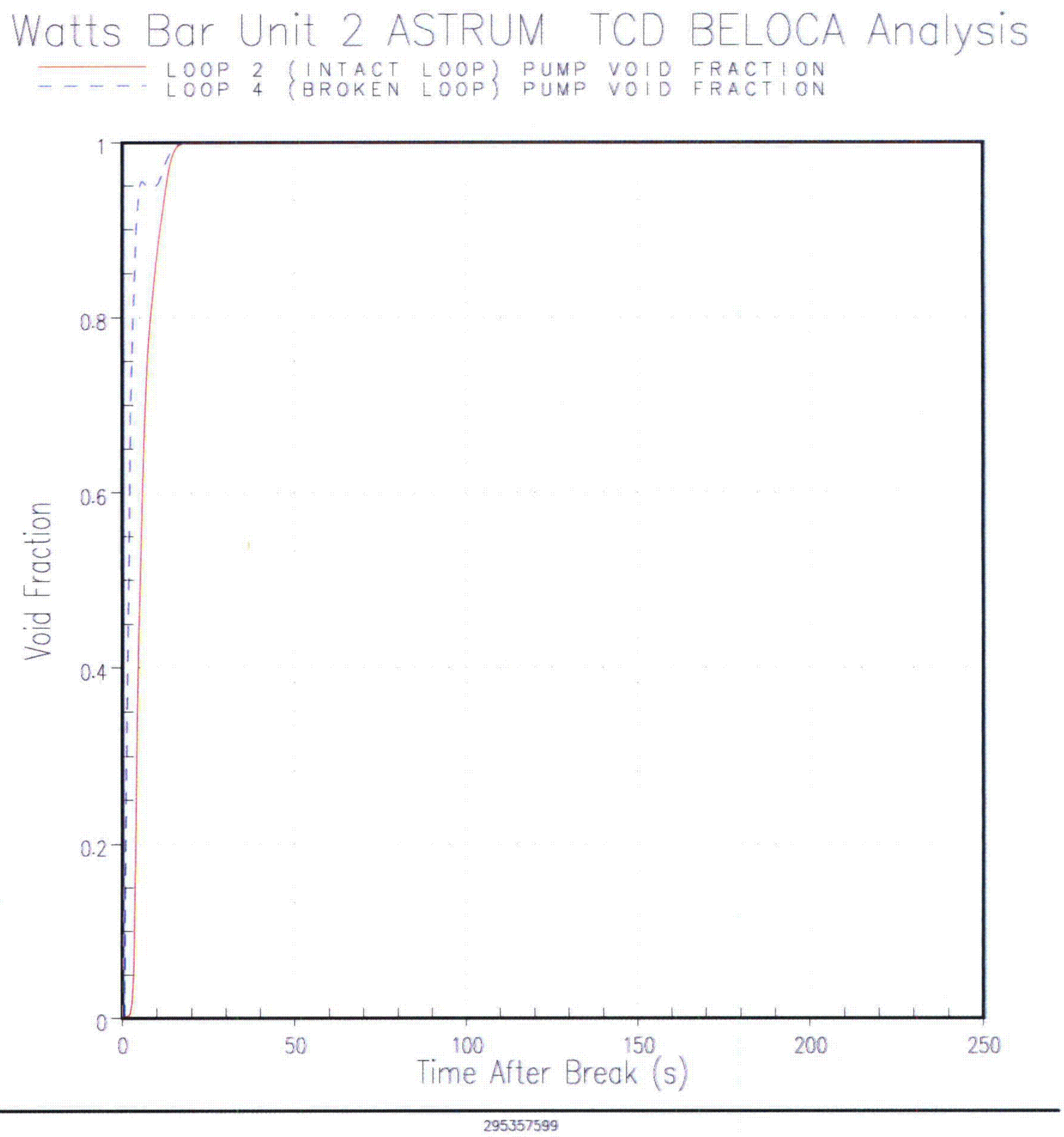

Figure 4
Watts Bar Unit 2 Limiting PCT Case Broken and Intact Loop Pump Void Fractions

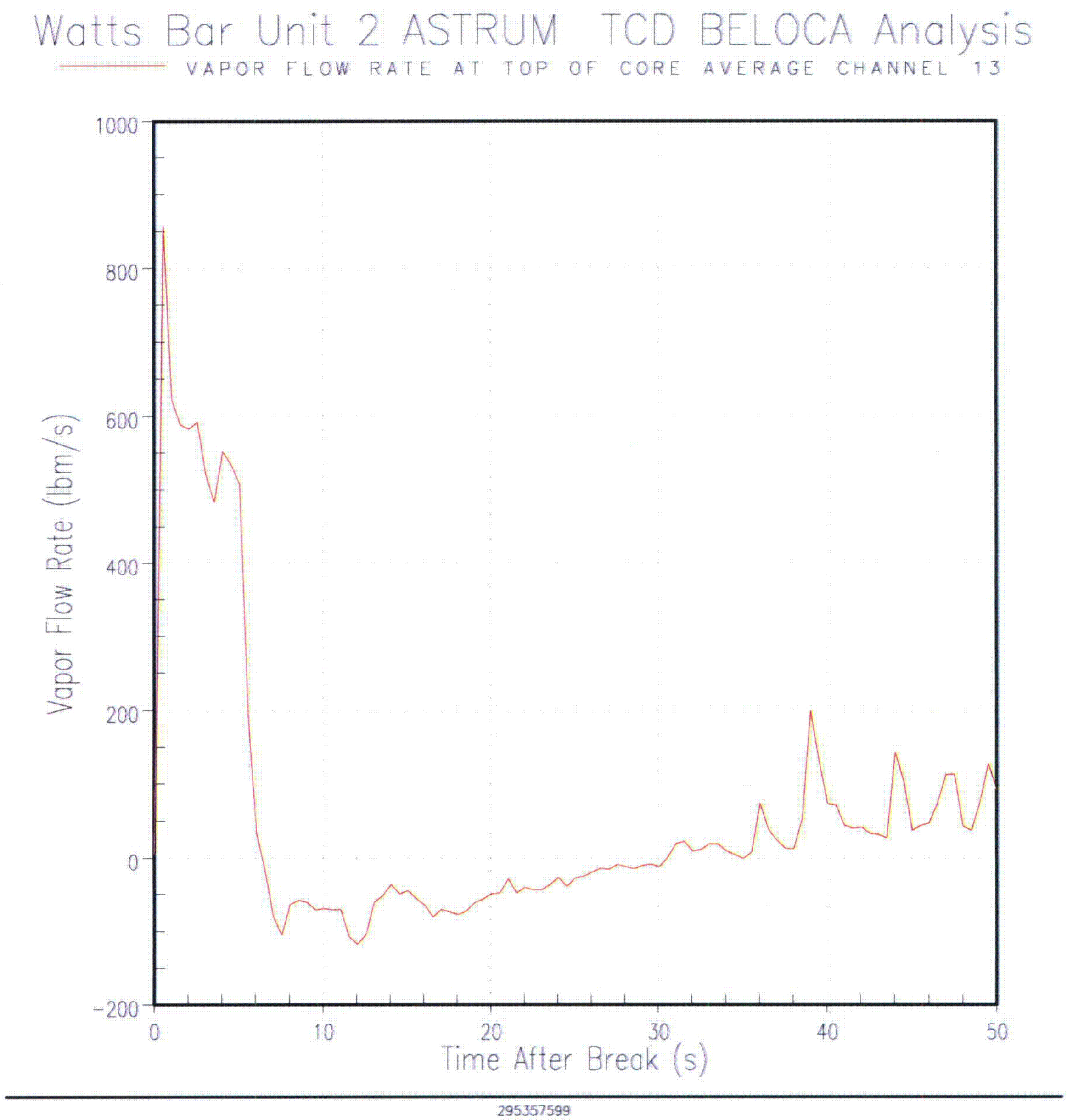

Figure 5
Watts Bar Unit 2 Limiting PCT Case Core Vapor Flow at the Top of the Core for a Core Average Channel

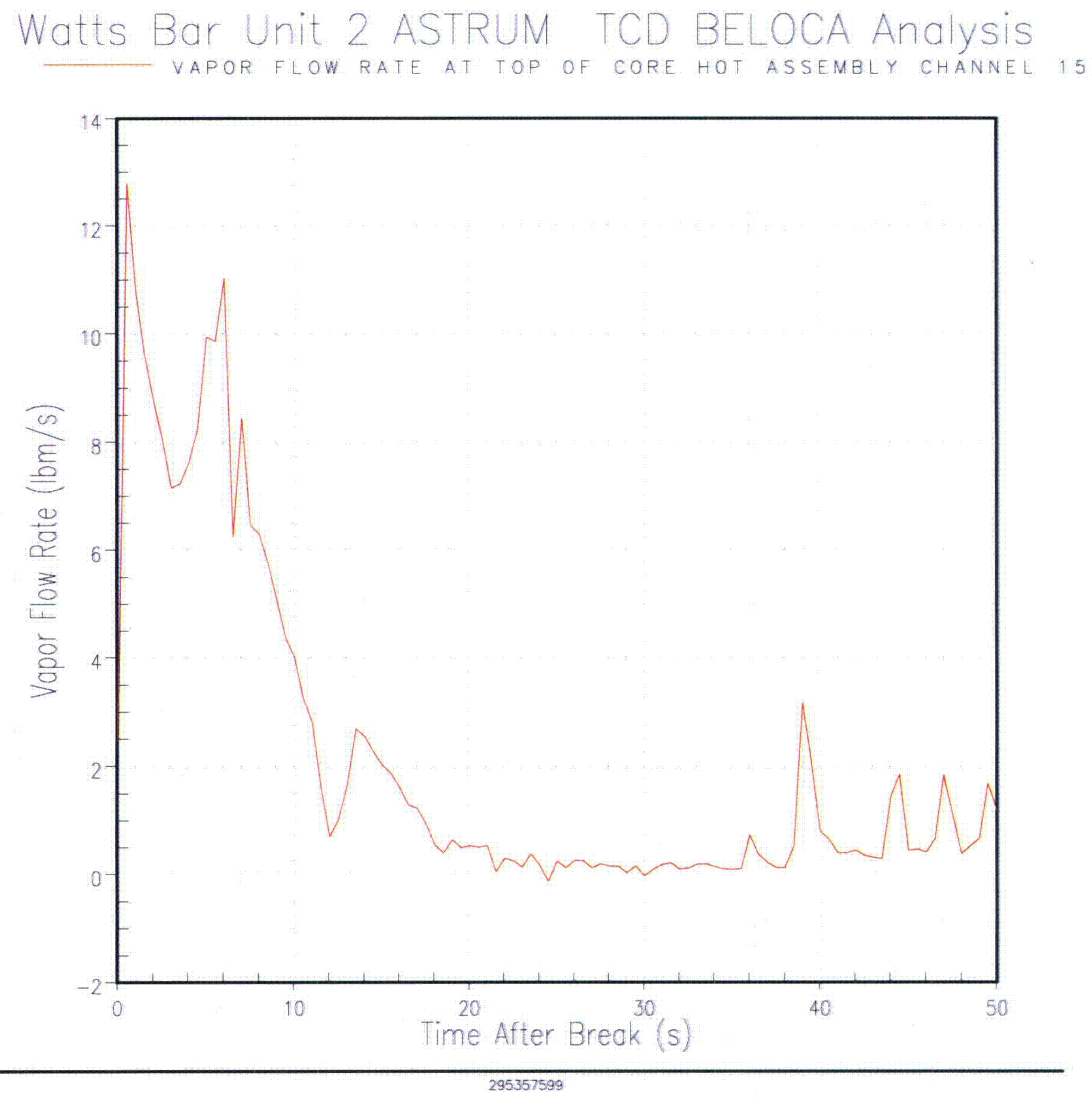

Figure 6
Watts Bar Unit 2 Limiting PCT Case Core Vapor Flow at the Top of the Core for the Hot Assembly Channel

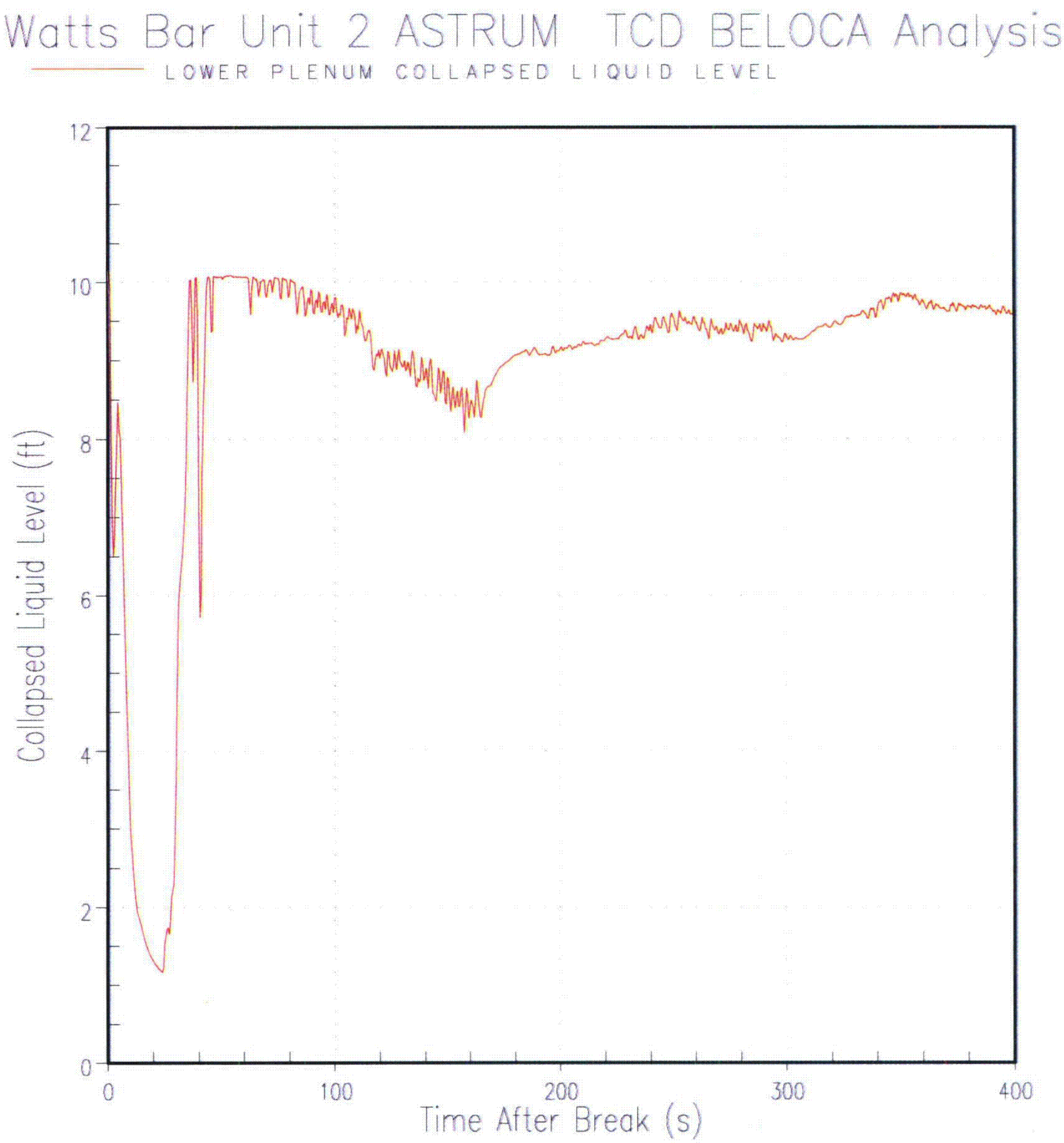

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Figure 7
Watts Bar Unit 2 Limiting PCT Case Lower Plenum Collapsed Liquid Level

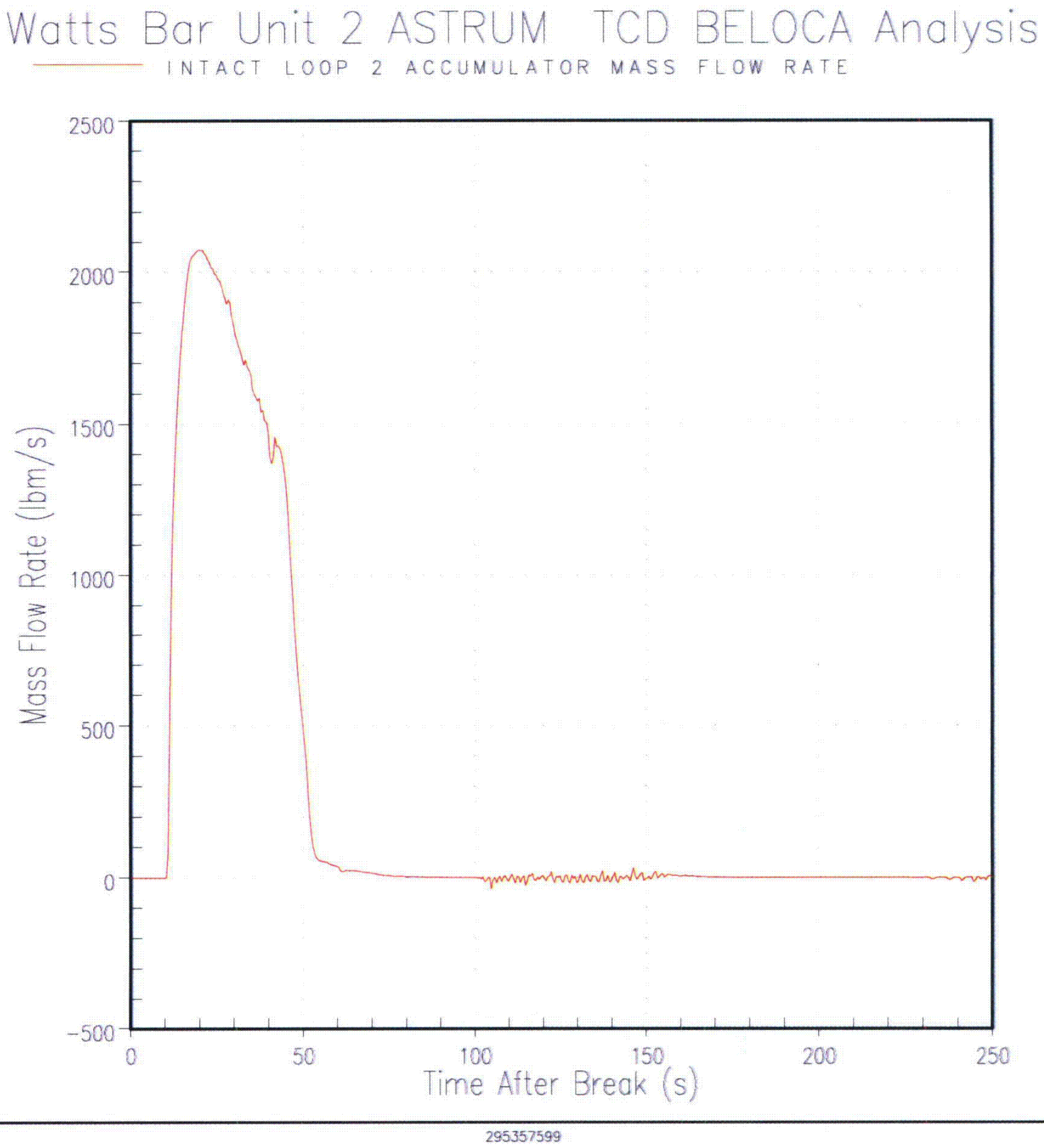

Figure 8
Watts Bar Unit 2 Limiting PCT Case Intact Loop Accumulator Flow

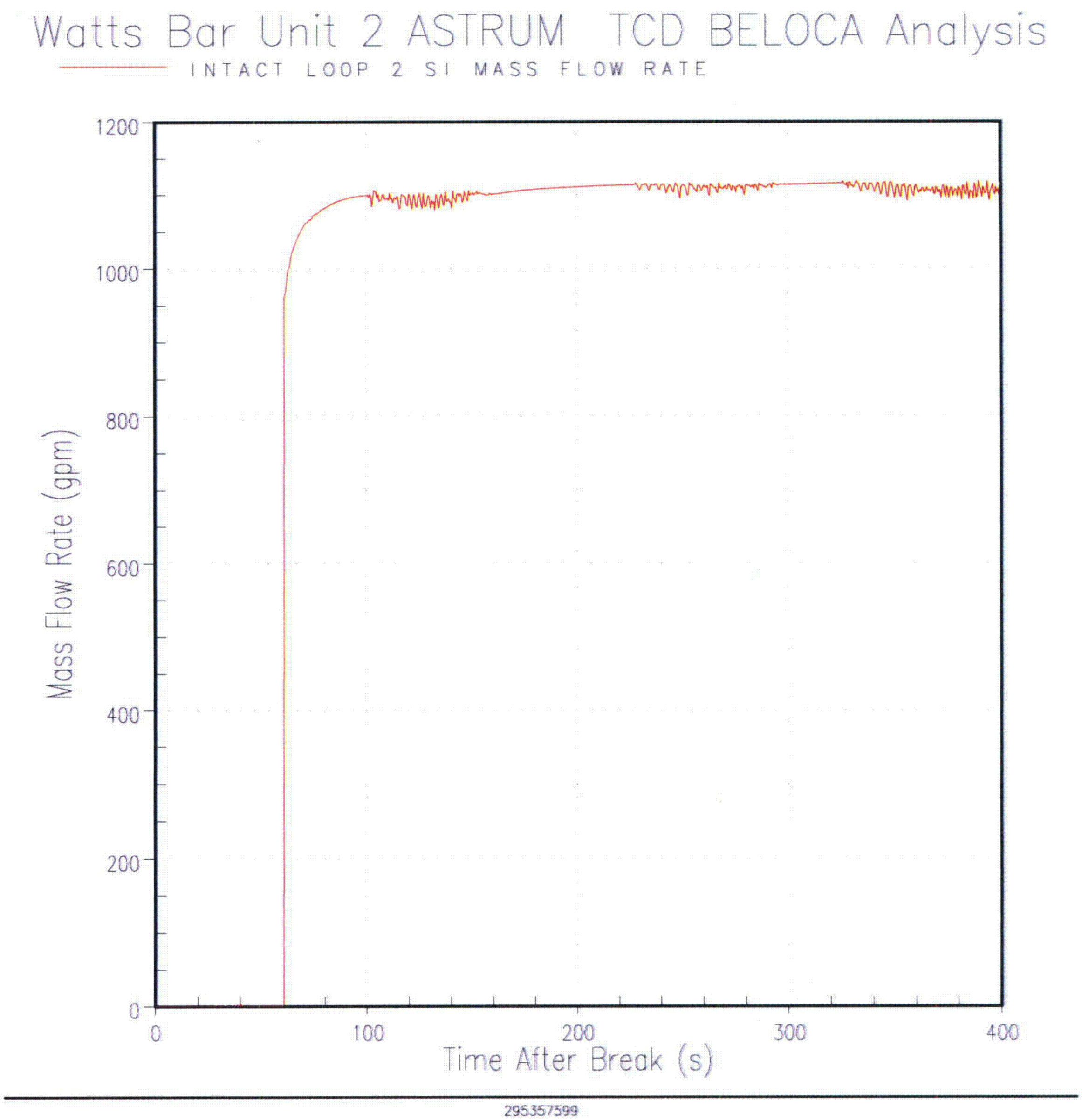

Figure 9
Watts Bar Unit 2 Limiting PCT Case Intact Loop Safety Injection Flow

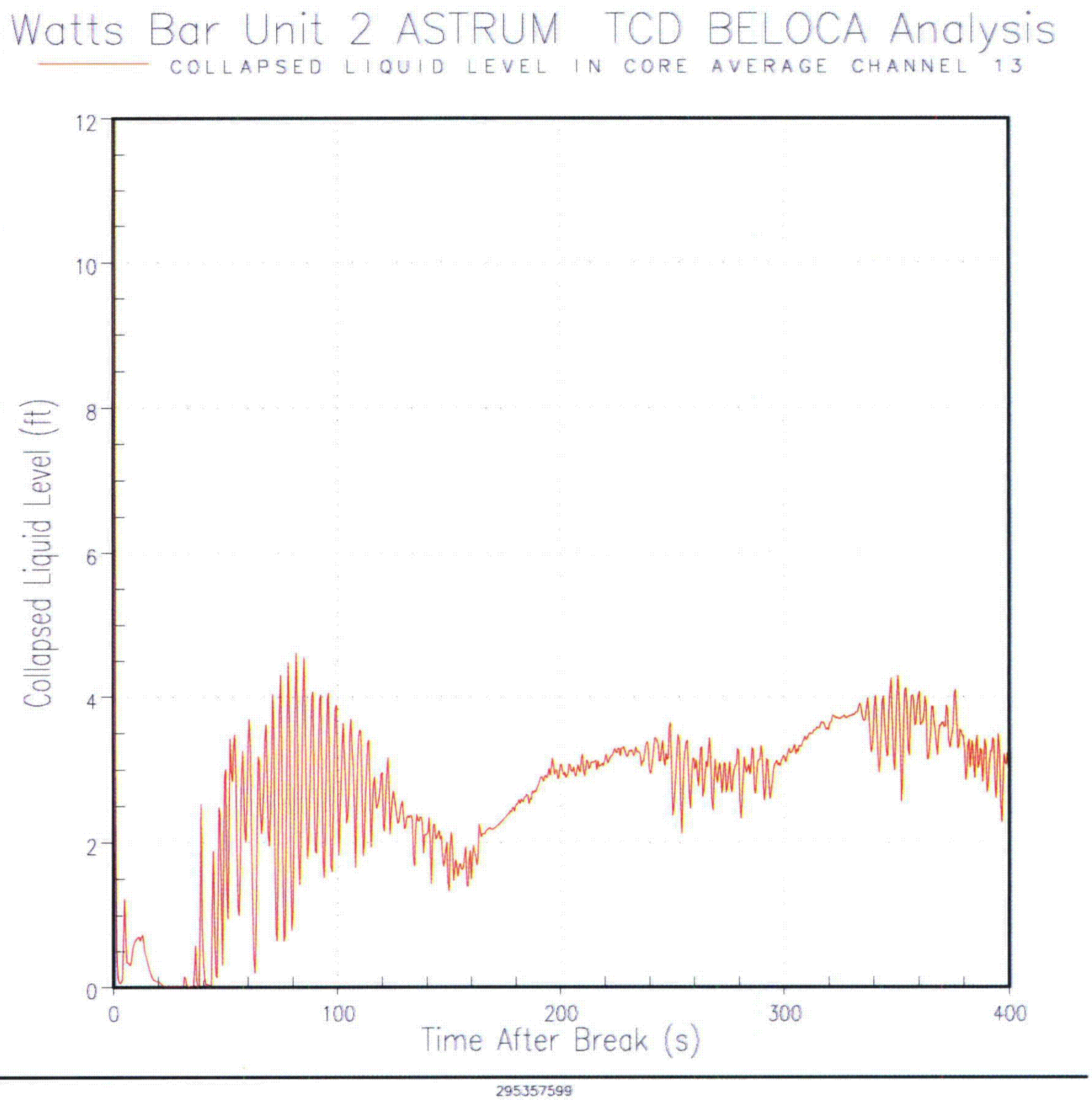

Figure 10
Watts Bar Unit 2 Limiting PCT Case Core Average Channel Collapsed Liquid Level

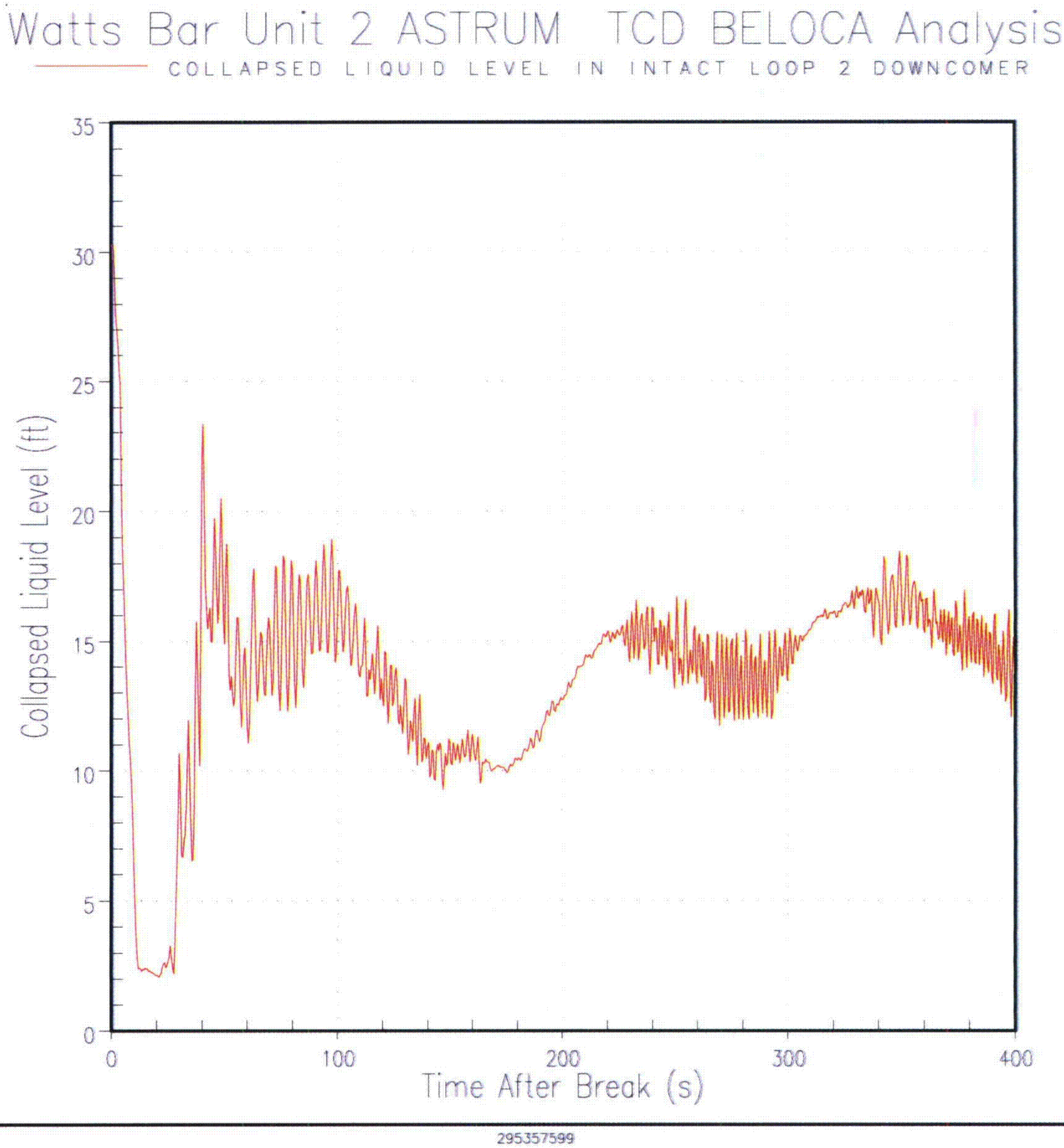

Figure 11
Watts Bar Unit 2 Limiting PCT Case Loop 2 Downcomer Collapsed Liquid Level

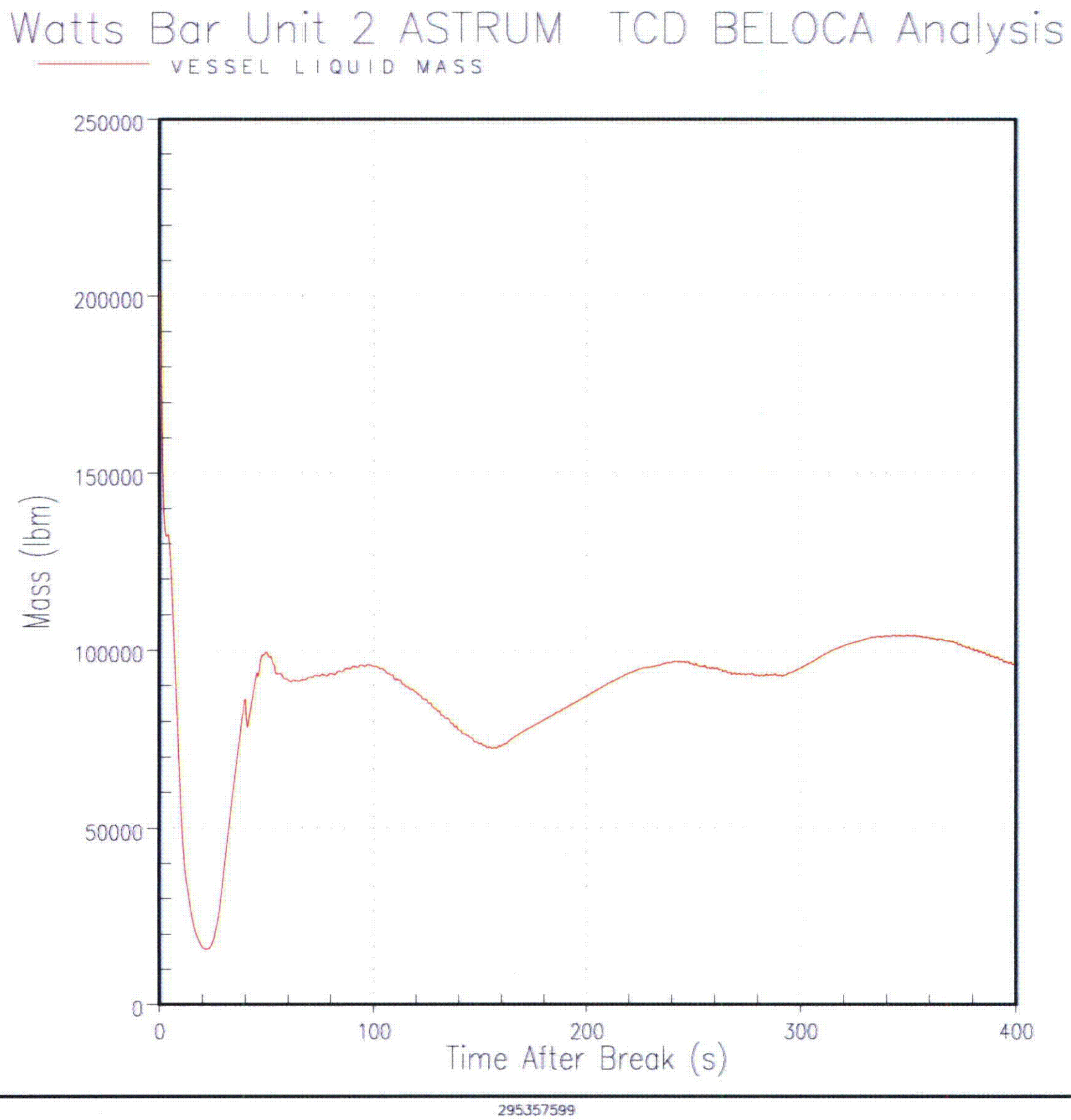

Figure 12
Watts Bar Unit 2 Limiting PCT Case Vessel Fluid Mass

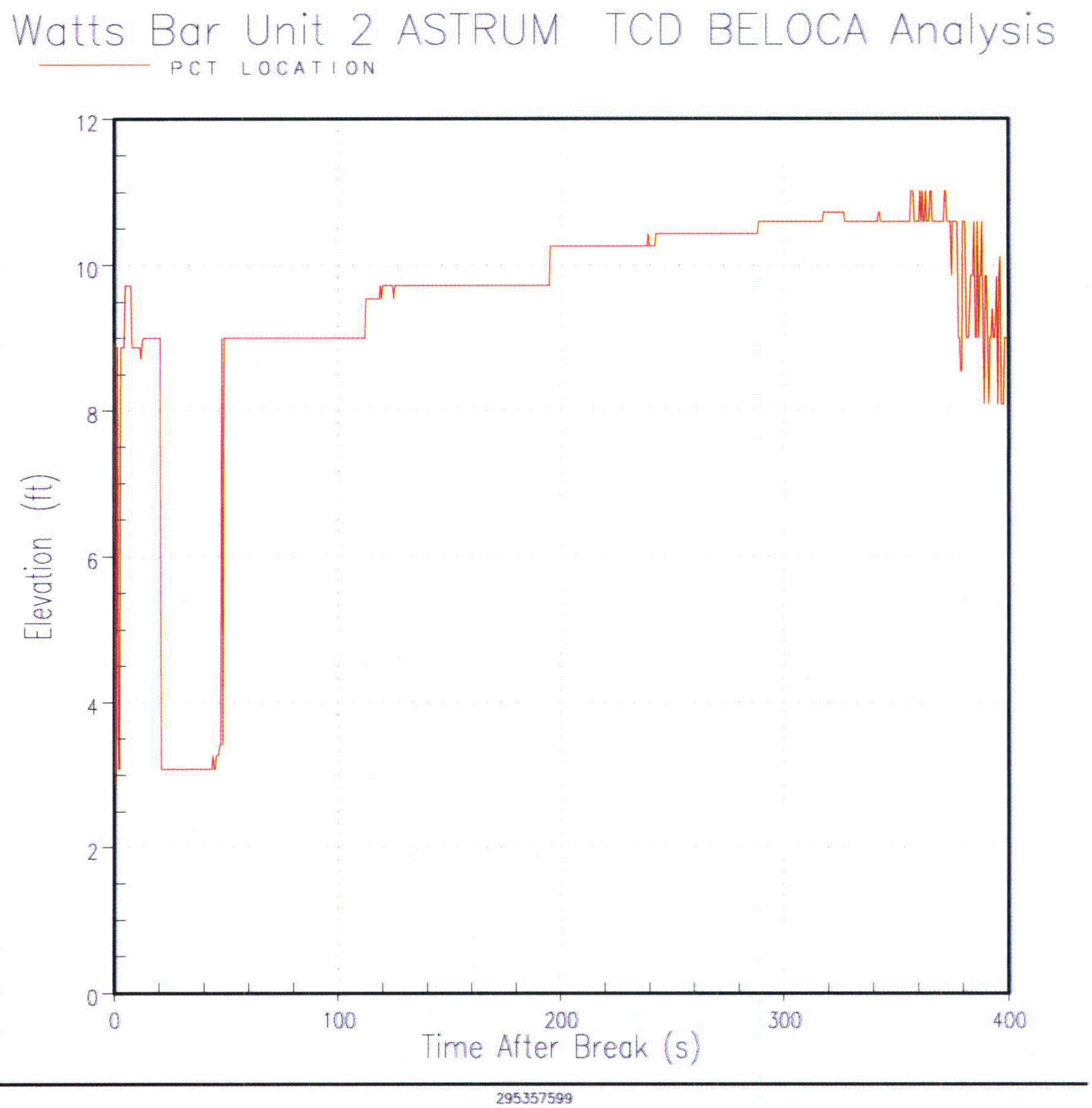

Figure 13
Watts Bar Unit 2 Limiting PCT Case PCT Location

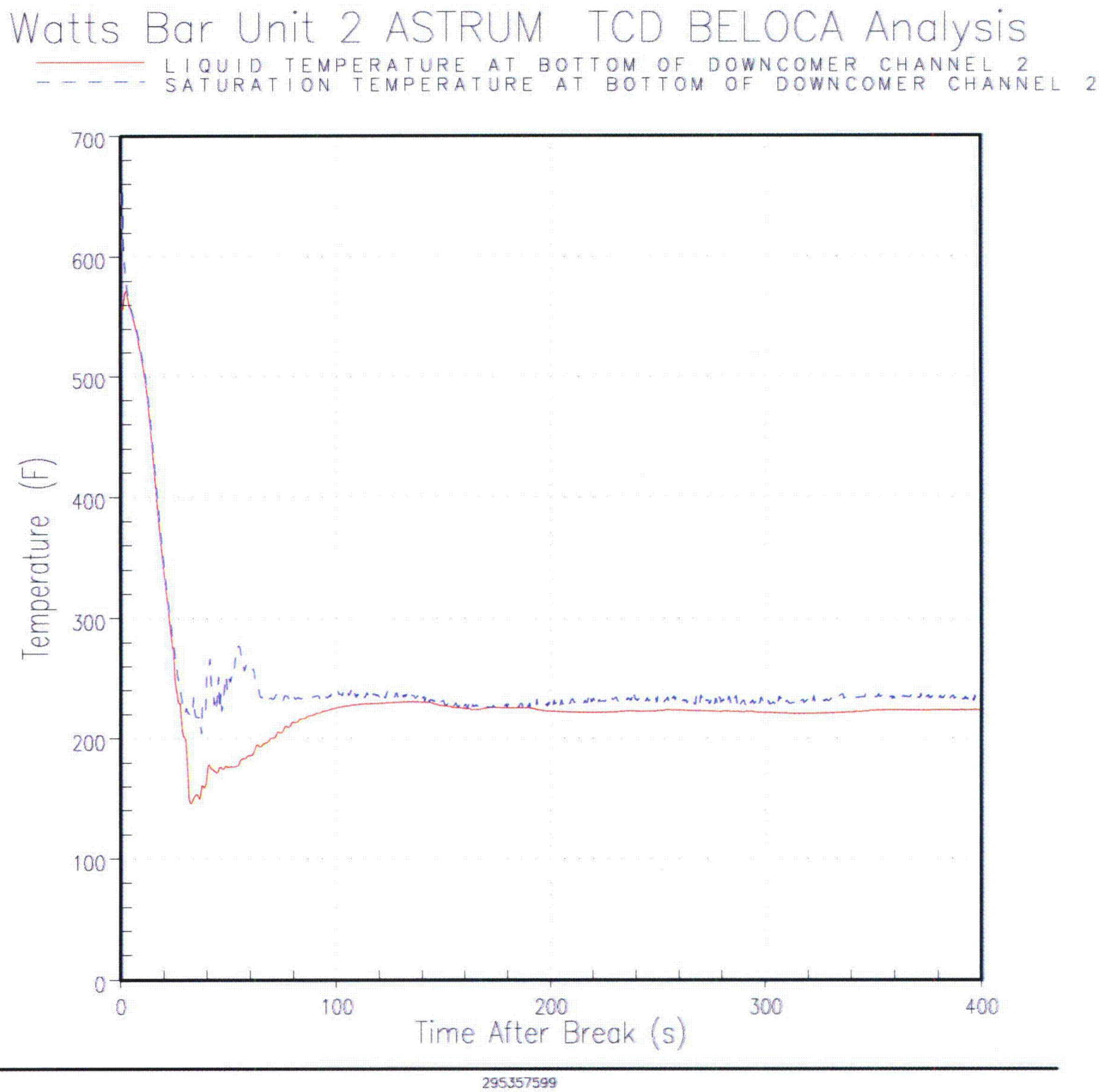

Figure 14
Watts Bar Unit 2 Limiting PCT Case Liquid and Saturation Temperature at Bottom of Downcomer Channel

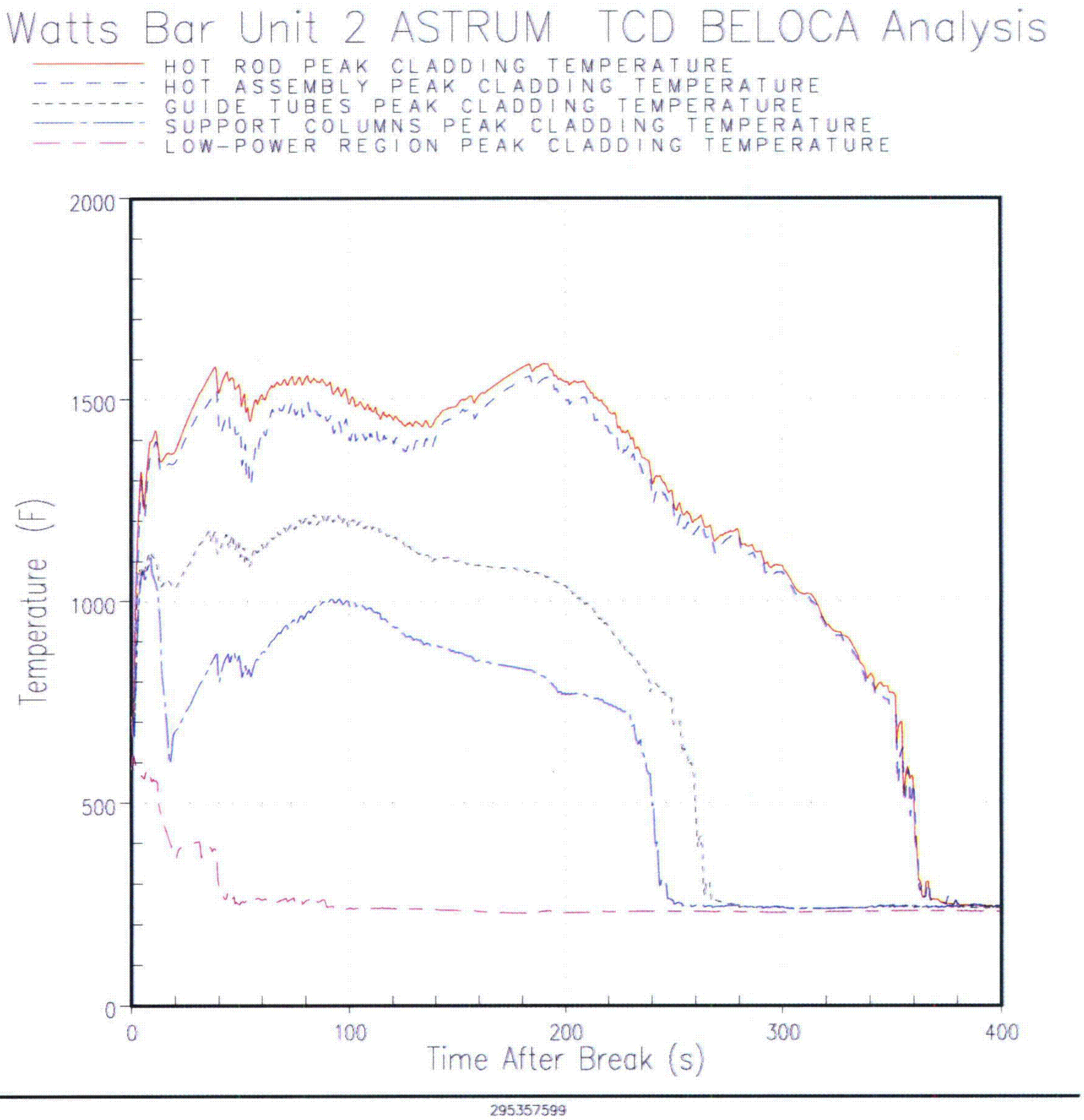

Figure 15
Watts Bar Unit 2 Limiting PCT Case Peak Cladding Temperature for all 5 Rods

ENCLOSURE 4
Tennessee Valley Authority
Watts Bar Nuclear Plant, Unit 2
Docket No. 50-391

WBN UNIT 2 FSAR CHAPTER 15 MARK-UP

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15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR Part 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system (ECCS) and the containment. For the purposes of this report the following faults have been classified in this category:

(1) Major rupture of pipes containing reactor coolant up to and including double ended rupture of the largest pipe in the reactor coolant system (loss of coolant accident).

(2) Major secondary system pipe ruptures.

(3) Steam generator tube rupture.

(4) Single reactor coolant pump locked rotor.

(5) Fuel handling accident.

(6) Rupture of a control rod drive mechanism housing (rod cluster control assembly ejection).

The analysis of thyroid and whole body doses, resulting from events leading to fission product release, appears in Section 15.5. The fission product inventories which form a basis for these calculations are presented in Chapter 11 and Section 15.1. Section 15.5 also includes the discussion of systems interdependency contributing to limiting fission product leakages from the containment following a Condition IV occurrence.

15.4.1 Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Loss-of-coolant accidents (LOCAs) are accidents that would result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system. LOCAs could occur from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system (RCS). Large breaks are defined as breaks in the reactor coolant pressure boundary having a cross-sectional area greater than or equal to 1.0 ft^2. Reference [34] documents this criterion. The large break LOCA analysis is performed to demonstrate compliance with the 10 CFR 50.46 acceptance criteria[35] for emergency core cooling systems for light water nuclear power reactors.

A large break LOCA is the postulated double-ended guillotine or split rupture of one of the RCS primary coolant pipes.

The boundary considered for loss of coolant accidents is the RCS or any line connected to the system up to the first closed valve.

The sequence of events following a limiting large break LOCA transient is presented in Table 15.4-17. Before the break occurs, the RCS is assumed to be operating normally at full power in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. A large beak is assumed to open almost instantaneously in one of the main RCS pipes. Calculations have demonstrated that the most severe transient results occur for a break in the cold leg between the pump and the reactor vessel.

Immediately following the cold leg break, a rapid system depressurization occurs along with a core flow reversal due to a high discharge of subcooled fluid into the broken cold leg and out the break. The fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up, while the core power shuts down due to voiding in the core. The hot water in the core, upper plenum, and upper head flashes to steam, and subsequently the cooler water in the lower plenum and downcomer begins to flash. Once the system has depressurized to the accumulator pressure, the accumulators begin to inject cold borated water into the intact cold legs. During the blowdown period a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. The bypass period ends as the system pressure continues to decrease and approaches the containment pressure, resulting in reduced break flow and consequently reduced core flow.

As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water. This phase continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

During the reflood period, the core flow is oscillatory as ECCS water periodically rewets and quenches the hot fuel cladding which generates steam and causes system repressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out the break. This flow path resistance is overcome by the downcomer water elevation head which provides the gravity driven reflood force. The pumped ECCS water aids in the filling of the downcomer and subsequently supplies water to maintain a full downcomer and complete the reflood period.

15.4.1.1 Thermal Analysis

15.4.1.1.1 Westinghouse Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss of coolant accident including the double ended severance of the largest reactor coolant system pipe. The reactor core and internals together are designed so that the reactor can be safely shutdown and the essential heat transfer geometry of the core preserved following the accident. The current internals is of the upflow barrel/baffle design. The

ECCS, even when operating during the injection mode with the most limiting single active failure, is designed to meet the acceptance criteria.

15.4.1.1.2 Method of Thermal Analysis

When the Final Acceptance Criteria (FAC) governing the loss-of-coolant accident (LOCA) for Light Water Reactors was issued in Appendix K of 10 CFR 50.46, both the Nuclear Regulatory Commission (NRC) and the industry recognized that the stipulations of Appendix K were highly conservative. That is, using the then accepted analysis methods, the performance of the Emergency Core Cooling System (ECCS) would be conservatively underestimated, resulting in predicted Peak Clad Temperatures (PCTs) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1) Identify, through separate effects and integral effects experiments, the degree of conservatism in those models permitted in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed.

2) Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without loss in safety to the public. It was confirmed that some plants were being restricted in operating flexibility by the overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472 [50]. The SECY-83-472 [50] represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1998, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models", to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best-estimate codes is provided in Regulatory Guide 1.157[44].

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (NUREG/CR-5249[45]). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis. A LOCA evaluation methodology for three- and four-loop Pressurized Water Reactor (PWR) plants based on the revised10 CFR 50.46 rules was developed by Westinghouse with support of EPRI and Consolidated Edison and has been approved by the NRC (WCAP-12945-P-A [46]).

More recently, Westinghouse developed an alternative methodology called ASTRUM, which stands for Automated Statistical TReament of Uncertainty Method (WCAP-16009-P-A [49]). This method is still based on the CQD methodology and follows the steps in the CSAU methodology (NUREG/CR-5249 [45]). However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in WCAP-16009-P-A [49].

The three 10 CFR 50.46 criteria (peak clad temperature, maximum local oxidation, and core-wide oxidation) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size). In particular, the statistical theory predicts that 124 calculations are required to simultaneously bound the 95th percentile values of three parameters with a 95-percent confidence level.

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A [49], as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A [49] was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and CQD uncertainty approach per section 4.0 of the ASTRUM Final Safety Evaluation Report appended to this topical report. Insert A

The methods used in the application of WCOBRA /TRAC to the large break LOCA with ASTRUM are described in WCAP-12945-P-A [46] and WCAP-16009-P-A [49]. A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR large break LOCA. Modeling of a PWR introduces additional uncertainties which are identified and quantified in the plant-specific analysis. ~~WCOBRA/TRAC MOD7A was used for the execution of ASTRUM for Watts Bar Unit 2 (WCAP-16009-P-A [49]).~~

Insert B

WCOBRA/TRAC combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

1) Ability to model transient three-dimensional flows in different geometries inside the 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, reactor coolant pumps, etc.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ. Dividing the liquid phase into two fields is a convenient and physically accurate way of handling flows where the liquid can appear in both film and droplet form. The droplet field permits more accurate modeling of thermal-hydraulic phenomena such as entrainment, de-entrainment, fallback, liquid pooling, and flooding.

WCOBRA/TRAC also features a two-phase, one-dimensional hydrodynamic formulation. In this model, the effect of a phase slip is modeled indirectly via a constitutive relationship which provides the phase relative velocity as a function of fluid conditions. Separate mass and energy conservation equations exist for the two-phase mixture and for the vapor.

The reactor vessel is modeled with the three-dimensional, three-field model, while the loop, major loop components, and safety injection points are modeled with the one-dimensional model.

All geometries modeled using the three-dimensional model are represented as a matrix of cells. The number of mesh cells used depends on the degree of detail required to resolve the flow field, the phenomena being modeled, and practical restrictions such as computing costs and core storage limitations.

The equations for the flow field in the three-dimensional model are solved using a staggered difference scheme on the Eulerian mesh. The velocities are obtained at mesh cell faces, and the state variables (e.g., pressure, density, enthalpy, and phasic volume fractions) are obtained at the cell center. This cell is the control volume for the scalar continuity and energy equations. The momentum equations are solved on a staggered mesh with the momentum cell centered on the scalar cell face.

The basic building block for the mesh is the channel, a vertical stack of single mesh cells. Several channels can be connected together by gaps to model a region of the reactor vessel. Regions that occupy the same level form a section of the vessel.

Vessel sections are connected axially to complete the vessel mesh by specifying channel connections between sections. Heat transfer surfaces and solid structures that interact significantly with the fluid can be modeled with rods and unheated conductors.

One-dimensional components are connected to the vessel. The basic scheme used also employs the staggered mesh cell. Special purpose components exist to model specific components such as the steam generator and pump.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood proceeds continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the LOTIC-2 [5] code and mass and energy releases from the WCOBRA/TRAC calculation .

The final step of the best-estimate methodology, in which all uncertainties of the LOCA parameters are accounted for to estimate a Peak Cladding Temperature (PCT), Maximum Local Oxidation (MLO), and Core-Wide Oxidation (CWO) at 95-percent probability, is described in the following sections.

1) Plant Model Development:

   In this step, a WCOBRA/TRAC model of the plant is developed. A high level of noding detail is used in order to provide an accurate simulation of the transient. However, specific guidelines are followed to ensure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences, such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

2) Determination of Plant Operating Conditions:

   In this step, the expected or desired operating range of the plant to which the analysis applies is established. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model.

   A transient is run utilizing these parameters and and is known as the "initial transient". Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. Because certain parameters are not included in the uncertainty analysis, these parameters are set at their bounding condition. This analysis is commonly referred to as the confirmatory analysis. The most limiting input conditions, based on these confirmatory runs, are then combined into the model that will represent the limiting state for the plant, which is the starting point for the assessment of uncertainties.

Insert C

3) Assessment of Uncertainty:

The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of WCAP-16009-P-A [49]. The determination of the PCT uncertainty, MLO uncertainty, and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA /TRAC calculations are necessary to assess against the three 10 CFR 50.46 criteria (PCT,MLO,CWO).

The uncertainty contributors are sampled randomly from their respective distributions for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each time in the cycle, break type (split or double-ended guillotine), and break size for the split break are also sampled as uncertainty contributors within the ASTRUM methodology.

Results from the 124 calculations are tallied by ranking the PCT from highest to lowest. A similar procedure is repeated for MLO and CWO. The highest rank of PCT, MLO, and CWO will bound 95 percent of their respective populations with 95-percent confidence level.

4) Plant Operating Range:

The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, this range may be the desired range or may be narrower for some parameters to gain additional margin.

15.4.1.1.3 Containment Analysis

The containment pressure analysis is performed with the LOTIC-2 [5] code. Transient mass and energy releases for input to the LOTIC-2 model are obtained from the WCOBRA/TRAC code. The transient pressure computed by the LOTIC-2 code is then used in WCOBRA/TRAC for the purpose of supplying a backpressure at the break plane while computing the reflood transient. The containment pressure transients and associated parameters were computed by LOTIC-2 and are presented in Figures 15.4-40b through 15.4-40g. The data used to model the containment for the analysis is presented in Tables15.4-14 and 15.4-15. Mass and energy release rates to containment can be found in Table 15.4-16.

Insert D

The impact of purging on the calculated containment pressure was addressed by performing a calculation to obtain the amount of mass which exits through two available purge lines during the initial portion of a postulated LOCA transient. The maximum air loss was calculated using the transient mass distribution (TMD) computer code model, which is described in Section 6.2.1.3.4, to be 1160 lbm. The containment pressure calculations account for a loss of 1160 lbm of air after initiation of the accident through modifying the compression ratio input to the LOTIC-2 code.

15.4.1.1.4 Results of Large Break Limiting Transient

The Watts Bar Unit 2 ~~PCT/MLO/CWO - limiting transient is a cold leg split break (effective break area = 1.8138)~~ which analyzes conditions that fall within those listed in Table 15.4-19. Traditionally, cold leg breaks have been limiting for large break LOCA. Analysis experience indicates that this break location most likely causes conditions that result in flow stagnation to occur in the core. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (WCAP-12945-P-A[46]).

Insert E

The large break LOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cool down transients. For a typical large break, the blowdown period can be divided into the Critical Heat Flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long-term cooling periods. Specific important transient phenomena and heat transfer regimes are discussed below, with the transient results shown in Figure 15.4-41 through 15.4-55. (The limiting case was chosen to show a conservative representation of the response to a large break LOCA.)

PCT

Figure 15.4-41

1) Critical Heat Flux (CHF) Phase:

Immediately following the cold leg rupture, the break discharge rate is subcooled and high (Figure 15.4-42). The region of the RCS with the highest initial temperatures (core, upper plenum, upper head, and hot legs) begin to flash to steam, the core flow reverses and the fuel rods begin to go through departure from nucleate boiling (DNB). The fuel cladding rapidly heats up (~~Figures 15.4-41a and 15.4-41b~~) while the core power shuts down due to voiding in the core. This phase is terminated when the water in the lower plenum and downcomer begins to flash (Figures 15.4-47 and 15.4-51). The mixture swells and intact loop pumps, still rotating in single phase liquid, push this two-phase mixture into the core.

2) Upward Core Flow Phase:

Heat transfer is improved as the two-phase mixture is pushed into the core. This phase may be enhanced if the pumps are not degraded, or if the break discharge rate is low due to saturated fluid conditions at the break. If pump degradation is high or the break flow is large, the cooling effect due to upward flow may not be significant. Figure 15.4-44 shows the void fraction for one intact loop pump and the broken loop pump. This figure shows that the intact loop remains in single-phase liquid flow for several seconds, resulting in enhanced upward core flow cooling. This phase ends as the lower plenum mass is depleted, the loop flow becomes two-phase, and the pump head degrades.

3) Downward Core Flow Phase:

The loop flow is pushed into the vessel by the intact loop pumps and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core, up the downcomer to the broken loop cold leg, and out the break. While liquid and entrained liquid flow provide core cooling, the top of the core vapor flow (Figures 15.4-45 and 15.4-46) best illustrates this phase of core cooling. Once the system has depressurized to the accumulator pressure (Figure 15.4-43), the accumulators begin to inject cold borated water into the intact cold legs (Figure 15.4-48). During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. As the system pressure continues to fall, the break flow, and consequently the downward core flow, is reduced. The core begins to heat up as the system pressure approaches the containment pressure and the vessel begins to fill with ECCS water (Figure 15.4-52).

4) Refill Phase:

As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water (Figure~~s~~ 15.4-48 ~~and 15.4-49~~). This period is characterized by a rapid increase in cladding temperatures at all elevations due to the lack of liquid and steam flow in the core region. This period continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

5) Early Reflood Phase:

(Figure 15.4-49)

During the early reflood phase, the accumulators begin to empty and nitrogen enters the system. This forces water into the core, which then boils, causing system re-pressurization, and the lower core region begins to quench (Figure 15.4-50). During this time, core cooling may increase due to vapor generation and liquid entrainment. During the reflood period, the core flow is oscillatory as cold water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. From the later stage of blowdown to the beginning of reflood, the accumulators rapidly discharge borated cooling water into the RCS, filling the lower plenum and contributing to the filling of the downcomer. The pumped ECCS water aids in the filling of the downcomer and subsequently supplies water to maintain a full downcomer and complete the reflood period. As the quench front progresses up the core, the PCT location moves higher into the top core region. As the vessel continues to fill, the PCT location is cooled and the early reflood period is terminated.

6) Late Reflood Phase:

The late reflood phase is characterized by boiling in the downcomer. The mixing of ECCS water with hot water ~~and~~ and steam from the core, in addition to the continued heat transfer from the hot vessel metal, reduces the subcooling of water in the lower

(Figure 15.4-41, HOTSPOT result)

plenum and downcomer. Figure 15.4-54 illustrates the reduction in lower plenum subcooling.

The saturation temperature is dictated by the containment backpressure. For WBN, which has a low containment pressure after the LOCA, boiling does occur and has a significant effect on the gravity reflood. Vapor generated in the downcomer reduces the driving head which results in a reduced core reflood rate. The top core elevations experience a second reflood heatup, which exceeds the first.

15.4.1.1.5 POST ANALYSIS OF RECORD EVALUATIONS

Insert F

In addition to the analyses presented in this section, evaluations and reanalyses may be performed as needed to address computer code errors and emergent issues, or to support plant changes. The issues or changes are evaluated, and the impact on the Peak Cladding Temperature (PCT) is determined. The resultant increase or decrease in PCT is applied to the analysis of record PCT. The PCT, including all penalties and benefits is presented in Tables 15.4-18a for the large break LOCA. The current PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200 °F.

In addition, 10 CFR 50.46 requires that licensees assess and report the effect of changes to or errors in the evaluation model used in the large break LOCA analysis. These reports constitute addenda to the analysis of record provided in the FSAR until overall changes become significant as defined by 10 CFR 50.46. If the assessed changes or errors in the evaluation model results in significant changes in calculated PCT, a schedule for formal reanalysis or other action as needed to show compliance will be addressed in the report to the NRC.

Finally, the criteria of 10 CFR 50.46 requires that holders and users of the evaluation models establish a number of definitions and processes for assessing changes in the models or their use. Westinghouse, in consultation with the PWR Owner's Group (PWROG), has developed an approach for compliance with the reporting requirements. This approach is documented in WCAP-13451 [36], Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting. TVA provides the NRC with annual and 30-day reports, as applicable, for Watts Bar Unit 2. TVA intends to provide future reports required by 10 CFR 50,46 consistent with the approach described in WCAP-13451.

15.4.1.1.6 CONCLUSIONS - THERMAL ANALYSIS

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile at the 95-percent confidence level. Figure 15.4-41~~a~~ shows the predicted HOTSPOT cladding temperature ~~transient at the PCT location~~ and the WCOBRA/TRAC PCT transient, both for the limiting PCT case. The HOTSPOT PCT plot includes local uncertainties applied to the Hot Rod~~. Figure 15.4-41b presents the WCOBRA/TRAC PCT transient predicted for the limiting PCT case. This figure~~, whereas the WCOBRA/TRAC PCT plot does not account for any local

uncertainties. Since the resulting HOTSPOT PCT for the limiting case is
1766 ~~1552~~°F, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F, is demonstrated. The results are shown in Table 15.4-18b.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting MLO for the limiting case ~~is 1.04~~ percent, the analysis confirms that 10 CFR
1.99 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 15.4-18b.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total maximum oxidation ~~is 0.0~~ percent. A detailed
0.08 CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A detailed CWO calculation is therefore not needed because the outcome will always be less than the HAR value. Since the resulting CWO ~~is 0.0~~ percent, the analysis confirms that 10 CFR 50.46
0.08 acceptance criterion (b)(3), i.e., "Core Wide Oxidation less than 1 percent", is demonstrated.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that the fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The approved methodology (WCAP-12945-P-A [46]) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 44 assemblies in the low-power channel. This situation has not been calculated to occur for Watts Bar Unit 2. Therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that the long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. While WCOBRA/TRAC is typically not run past full core quench, all base calculations are run well past PCT turnaround and past the point where increasing vessel inventories are calculated. The conditions at the end of the WCOBRA/TRAC calculations indicate that the transition to long term cooling is underway even before the entire core is quenched.

Based on the ASTRUM Analysis results (Table 15.4-18b), it is concluded that Watts Bar Unit 2 maintains a margin of safety to the limits prescribed by 10 CFR 50.46.

15.4.1.1.7 PLANT OPERATING RANGE

Scope for markup

The expected PCT and its uncertainty developed are valid for a range of plant operating conditions. The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Tables 15.4-19 summarizes the operating ranges as defined for the proposed operating conditions which are supported by the Best-Estimate LBLOCA analysis for Watts Bar Unit 2. Tables 15.4-14 and 15.4-15 summarize the LBLOCA containment data used for calculating containment pressure. If operation is maintained within these ranges, the LBLOCA results developed in this report using WCOBRA/TRAC are considered to be valid. Note that some of these parameters vary over their range during normal operation (accumulator temperature) and other ranges are fixed for a given operational condition (Tavg).

15.4.1.2 Hydrogen Production and Accumulation

Pursuant to NRC final rule as defined in 10 CFR 50.44 and Regulatory Guide 1.7, the new definition of design-basis LOCA hydrogen release eliminates requirements for hydrogen control systems for mitigation of releases. "All PWRs with ice condenser type containments must have the capability to control combustible gas generated from metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment structural integrity. The deliberate ignition systems provided to meet this existing combustible gas source term are capable of safely accommodating even greater amounts of combustible gas associated with even more severe core melt sequences that fail the reactor vessel and involve molten core-concrete interaction. Deliberate ignition systems, if available, generally consume the combustible gas before it reaches concentrations that can be detrimental to containment integrity." On the basis of this definition, no further analysis is required to support events considered to be outside the design basis. Deliberate ignition systems are described in FSAR Section 6.2.5

15.4.2 Major Secondary System Pipe Rupture

15.4.2.1 Major Rupture of a Main Steam Line

15.4.2.1.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin.

released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.6.3 Conclusions

Even on a worst-case basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further, consequential damage to the reactor coolant system. The reference [16] analyses have demonstrated that the number of fuel rods entering DNB amounts to less than 10%, thus satisfactorily limiting fission product release.

The environmental consequences of this accident is bounded by the loss of coolant accident. See Section 15.5.3, "Environmental Consequences of a Loss of Coolant Accident." The reactor coolant system integrated break flow to containment following a rod ejection accident is shown in Figure 15.4-28.

Following reactor trip, requirements for operator action and protection system operation are similar to those presented in the analysis of a small loss of coolant event, section 15.3.1.

Scope for markup

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Table 15.4-14 Large-Break LOCA Containment Data (Ice Condenser Containment) Used for Calculation of Containment Pressure for Watts Bar Unit 2

Parameter	Value
Net Free Volume Distribution Between Upper (UC), Lower (LC), Ice Condenser (IC) and Dead-Ended (DE) Compartments	UC: 710,000 ft^3 LC: 253,114 ft^3 IC: 122,350 ft^3 DE: 129,900 ft^3
Initial Condition Containment Pressure	14.7 psia
Maximum Temperature for the Upper (UC), Lower (LC and Dead-Ended (DE) Compartments	UC: 110°F LC: 120°F DE: 120°F
Minimum RWST Temperature (Containment Spray Temperature)	60°F
Minimum Temperature Outside Containment	5°F
Maximum Containment Spray Flow Rate	4000 gpm/pump
Number of Spray Pumps Operating	2
Post-Accident Initiation of Spray System	25 sec
Post-Accident Delay Time for Deck Fan Actuation	490 sec
Deck Fan Flow Rate	41,690 cfm/fan
Initial Ice Mass	2,450,000 lb_m

Table 15.4-15 Large-Break Containment Data - Heat Sinks Data (Ice Condenser Containment)

Wall	Compartment(1)	Area [ft^2]	Thickness [ft]	Material
1	UC	5124.	1.6	concrete
2	UC	19992.	0.000525/1.6	coating/concrete
3	UC	4032.	0.02167/1.6	stainless steel/concrete
4	UC	11192.	0.00065/0.03908	coating/carbon steel
5	UC	47800.	0.00065/0.09252/1.0	coating/carbon steel/concrete
6	UC	273.	0.00065/0.1308	coating/carbon steel
7	LC	59000.	2.1	concrete
8	LC	17178.	0.000133/2.1	coating/concrete
9	LC	12988.	2.1	concrete
10	LC	2384.	0.02167/2.1	stainless steel/concrete
11	LC	25444.	0.00065/0.1089/1.0	coating/carbon steel/concrete
12	LC	12810.	0.00065/0.07593	coating/carbon steel
13	LC	2625.	0.00055/0.12083	coating/carbon steel
14	LC	1575.	0.00065/0.14167	coating/carbon steel
15	LC	12915.	0.00065/0.044167	coating/carbon steel
16	LC	12988.	2.1	concrete
17	LC	3439	0.1561	carbon steel

Notes:

1. UC and LC are Upper and Lower Compartment, respectively.

Table 15.4-16 Mass And Energy Release Rates Used for Calculation of Containment Pressure for Watts Bar Unit 2 (Page 1 of 2)

Time After Break (sec)	Mass Flow Rate(lbm/sec)	Energy Flow Rate (BTU/sec)
0.	9646.7	5369419.
1.	71201.5	39577048.
2.	50782.1	28747484.
3.	40475.1	23410743.
4.	34105.4	20560588.
5.	30009.1	18713303.
6.	27906.0	17640053.
7.	26130.9	16632257.
8.	24651.1	15663961.
9.	22805.6	14511306.
10.	20004.8	13053678.
11.	17472.9	11605252.
12.	14601.0	10093803.
12.4	13464.4	9420184.
14.	12172.5	7614137.
15.	12554.4	6455205.
16.	11369.7	5308157.
17.	10902.4	4491501.
18.	10124.7	3756484.
19.	9258.1	3127399.
20.	8178.7	2411114.
21.	7321.0	2120146.
22	7603.9	1977749.
23.	5474.9	1402837.
24.	4641.5	999621.
25.	6992.0	1356562.
26.	5955.4	1051498.
28.	4062.4	618361.
29.	3020.7	405978.
30.	1824.1	201868.
32.	1873.8	190499.

Table 15.4-16 Mass And Energy Release Rates Used for Calculation of Containment Pressure for Watts Bar Unit 2 (Page 2 of 2)

Time After Break (sec)	Mass Flow Rate(lbm/sec)	Energy Flow Rate (BTU/sec)
33.	1882.1	180047.
34.5	1890.2	204412.
35.	1921.9	200973.
39.	2275.7	246561.
41.	1959.7	214441.
43.	2031.8	267974.
45.	2650.2	384113.
46.	7824.1	1100908.
47.5	2842.5	400880.
50.	1811.6	373546.
51.	1764.3	397686.
55.	2254.1	544982.
57.5	1383.9	503576.
60.	1621.8	592463.
65.	790.9	338984.
80.	686.2	251517.
110.	646.9	232801.
150.	643.9	307300.
190.	654.1	229705.
226.	374.2	116811.
300.	404.3	144644.
349.	503.8	176903.

Table 15.4-17 Watts Bar Unit 2 Best-Estimate Large-Break LOCA Sequence Of Events for Limiting PCT Transient

Event	Time after break (sec)
Start of Transient	0.0
Safety Injection Signal	~~5.5~~ $\boxed{5}$
Accumulator Injection Begins	~~12.0~~ $\boxed{10}$
End of Blowdown	~~24.5~~ $\boxed{11}$
Bottom of Core Recovery	~~40.0~~ $\boxed{36}$
Accumulator Empty[1]	~~50.8~~ $\boxed{43}$
Safety Injection Begins	~~60.5~~ $\boxed{60}$
PCT Occurs	~~209.5~~ $\boxed{190}$
End of analysis time	400.0

Note:
1. Accumulator injection switches from liquid to nitrogen.

Table 15.4-18a Peak Clad Temperature Including All Penalties and Benefits, Best-Estmate Large-Break LOCA (BE LBLOCA) for Watts Bar Unit 2

PCT for Analysis-of-Record (AOR)	**~~1552~~°F** $\boxed{1766}$
PCT Assessments Allocated to AOR	
None	N/A
BE LBLOCA PCT for Comparison to 10 CFR 50.46 Requirements	~~1552~~°F $\boxed{1766}$

$\boxed{\text{AOR Value}}$

Table 15.4-18b Watts Bar Unit 2 Best-Estimate Large-Break LOCA Results

ASTRUM Results	~~Value~~	Acceptance Criteria
95/95 PCT	~~1552~~°F $\boxed{1766}$	<2200°F
95/95 MLO	~~1.04~~% $\boxed{1.99}$	<17%
95/95 CWO	~~0.0~~% $\boxed{0.08}$	<1%

Table 15.4-19 Plant Operating Range Analyzed by the Best-Estimate Large-Break LOCA Analysis for Watts Bar Unit 2 (Page 1 of 2)

	Parameter	As-Analyzed Value or Range
1.0	**Plant Physical Description**	
	a) Dimensions	Nominal
	b) Pressurizer location	Modeled on an intact loop
	c) Hot assembly location	Anywhere in core interior (1)
	d) Hot assembly type	17x17 RFA-2, ZIRLO® Clad with IFMs
	e) Steam generator tube plugging level	≤ 10% Any or All SGs
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Core Power	3479.8 MWt ±0% Uncertainty(2)
	b) Peak heat flux hot channel factor (F_Q)	≤2.50 $\boxed{\text{See Table 15.4-24}}$
	c) Peak hot rod enthalpy rise hot channel factor ($F_{\Delta H}$)	≤1.65 $\boxed{\text{See Table 15.4-24}}$
	d) Hot assembly radial peaking factor ($\overline{P}_{HA}$)	≤1.65/1.04 $\boxed{\text{See Table 15.4-24}}$
	e) Hot assembly heat flux hot channel factor (F_{QHA})	≤2.50/1.04 $\boxed{\text{See Table 15.4-24}}$
	f) Axial power distribution (P_{BOT},P_{MID})	Figure 15.4-56
	g) Low power region relative power (P_{LOW})	$0.2 \le P_{LOW} \le 0.8$
	h) Hot assembly burnup	≤ ~~75~~ $\boxed{62}$,000 MWD/MTU, lead rod ~~(3)~~
	i) MTC	≤ 0 at hot full power (HFP)
	j) Typical cycle length	20,000 MWD/MTU
	k) Minimum beginning of cycle core average burnup	≥ 10,000 MWD/MTU
	l) Maximum steady state depletion, F_Q	2.0 $\boxed{\text{See Table 15.4-24}}$
	2.2 Fluid Conditions	
	a) T_{AVG}	$582.2°F \le T_{AVG} \le 594.2°F$
	b) Pressurizer pressure	$2180 \text{ psia} \le P_{RCS} \le 2300 \text{ psia}$
	c) Loop flow	TDF ≥ 93,100 gpm/loop
	d) Upper head temperature	$= T_{COLD}$
	e) Pressurizer level (at full power)	1067 ft^3
	f) Accumulator temperature	$100°F \le T_{ACC} \le 120°F$
	g) Accumulator pressure	$585 \text{ psig} \le P_{AC} \le 690 \text{ psig}$
	h) Accumulator liquid volume	$1005 ft^3 \le V_{ACC} \le 1095 ft^3$

Table 15.4-19 Plant Operating Range Analyzed by the Best-Estimate Large-Break LOCA Analysis for Watts Bar Unit 2 (Page 2 of 2)

	Parameter	As-Analyzed Value or Range
	i) Accumulator fL/D	5.6186 ± 20%
	j) Minimum accumulator boron	1900 ppm (4)
3.0	**Accident Boundary Conditions**	
	a) Minimum safety injection flow	Table 15.4-23
	b) Safety injection temperature	60°F ≤ SI Temp ≤ 105°F
	c) Safety injection delay (5)	40 seconds (with offsite power) 55 seconds (with LOOP)
	d) Containment modeling	Tables 15.4-14, 15.4-15, and 15.4-16 and Figure 15.4-40b
	e) Single failure	1 RHR, 1 IHSI, and 1 CH/SI Pump Operable; Containment pressure: all trains operational

Notes:

1. 44 peripheral locations will not physically be lead power assembly.
2. The core average linear heat rate is set equal to a value corresponding to 3479.8 MWt (100.6 percent of 3459 MWt), and is not ranged in the uncertainty analysis. This power level approach bounds any future plant operation whose product of nominal full power and calorimetric uncertainty of ≤ 3479.8 MWt (for example, a nominal full power of 3479.8/1.005) MWt and 0.5% calorimetric uncertainty is bounded).
3. ~~Please note that the fuel temperature and rod internal pressure data is only provided up to 62,000 MWD/MTU. In addition, the hot assembly/hot rod will not have a burnup this high in ASTRUM analyses.~~

Not Used.

4. The accumulator boron concentration used for the uncertainty analysis was 1900 ppm rather than 3000 ppm, which was the value transmitted to Westinghouse by TVA. This bounds the value transmitted by TVA and will have no impact on the results presented herein.
5. Conservatively high SI delay times were used to bound the values transmitted by TVA to Westinghouse.

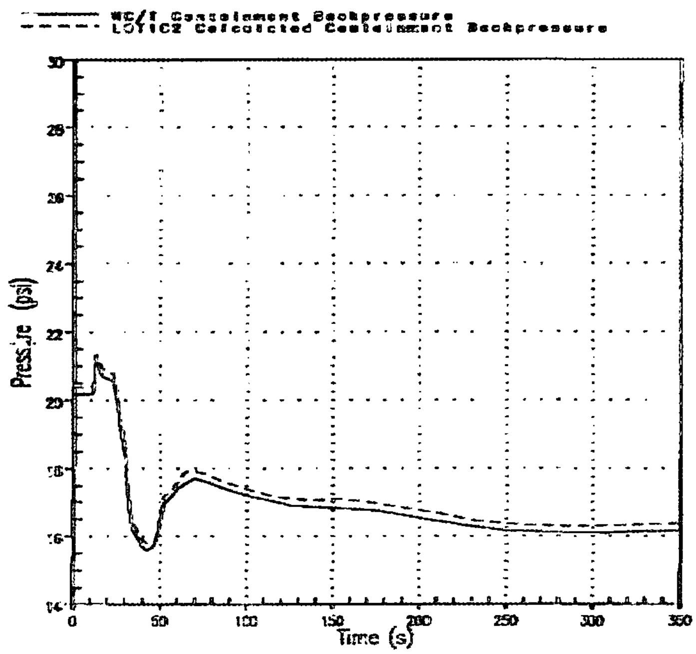
No Changes to Table 15.4-23

Table 15.4-23 Minimum Injected Safety Injection Flow Used in Best-Estimate Large-Break LOCA Analysis for Watts Bar Unit 2

Pressure [psia]	Charging Flow [gpm]	SI Flow [gpm]	RHR Flow [gpm]	Total Flow [gpm]
14.7	262.2	416.9	2715.2	3394.3
34.7	260.5	413.7	2284.6	2958.8
54.7	258.8	410.4	1811.3	2480.5
74.7	257.0	407.2	1367.7	2031.9
94.7	255.3	403.9	1156.9	1816.1
114.7	253.6	400.7	916.9	1571.2
134.7	251.9	396.8	633.2	1281.9
154.7	250.1	393.0	232.2	875.3
214.7	244.9	381.4	0.0	626.3
314.7	236.1	360.8	0.0	596.9
414.7	227.1	339.5	0.0	566.6
514.7	218.0	317.6	0.0	535.6
614.7	208.4	294.6	0.0	503.0
714.7	198.6	269.8	0.0	468.4
814.7	188.5	242.7	0.0	431.2
914.7	178.2	214.5	0.0	392.7
1014.7	167.6	185.0	0.0	352.6
1114.7	156.7	150.2	0.0	306.9
1214.7	145.4	150.9	0.0	251.3
1314.7	131.0	50.8	0.0	181.8
1414.7	115.9	0.0	0.0	115.9
1514.7	100.1	0.0	0.0	100.1
1614.7	83.0	0.0	0.0	83.0
1714.7	63.7	0.0	0.0	63.7
1814.7	44.3	0.0	0.0	44.3
1914.7	27.0	0.0	0.0	27.0
2014.7	5.8	0.0	0.0	5.8
2114.7	0.0	0.0	0.0	0.0

Table Insert 1

Figure 15.4-40a Deleted by Amendment 97



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
LOWER BOUND CONTAINMENT
PRESSURE
FIGURE 15.4-40b

Figure 15.4-40b Watts Bar Unit 2 Lower Bound Containment Pressure

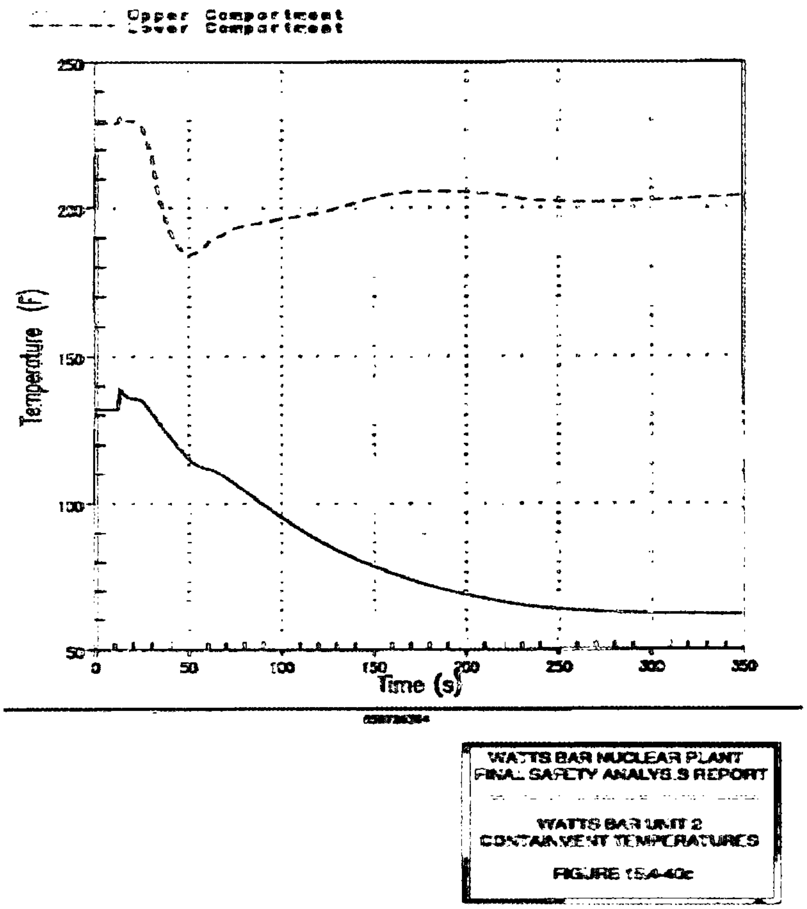
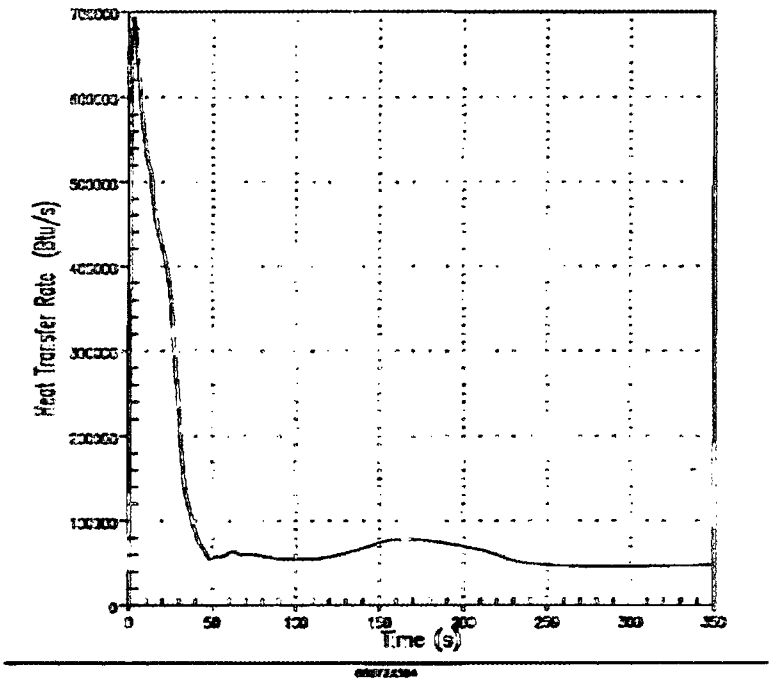


Figure 15.4-40c Watts Bar Unit 2 Containment Temperatures

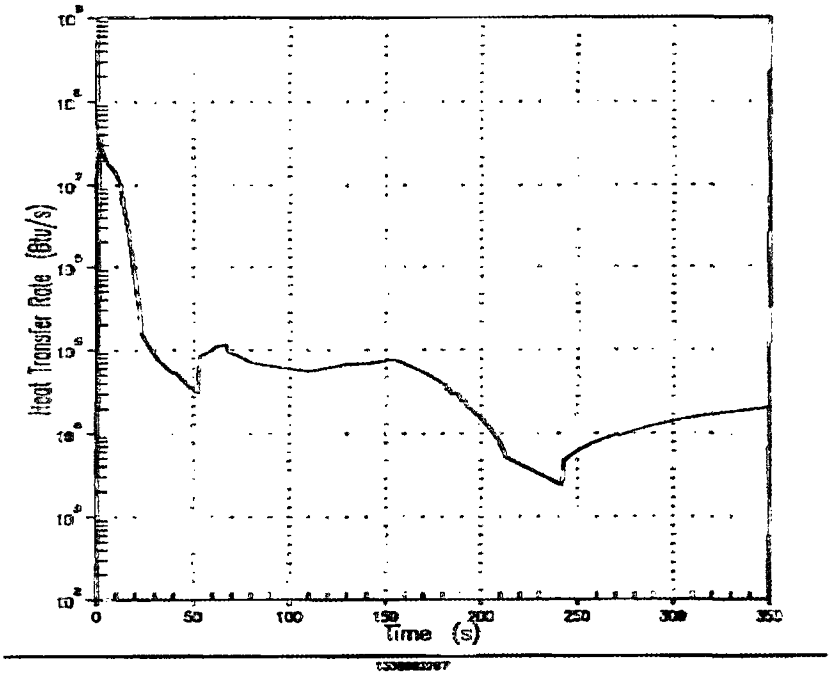


WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
LOWER COMPARTMENT
STRUCTURAL HEAT REMOVAL RATE

FIGURE 15.4-40d

Figure 15.4-40d Watts Bar Unit 2 Lower Compartment Structural Heat Removal Rate



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
WATTS BAR UNIT 2
ICE BED HEAT REMOVAL RATE
FIGURE 15.4-40e

Figure 15.4-40e Watts Bar Unit 2 Ice Bed Heat Removal Rate

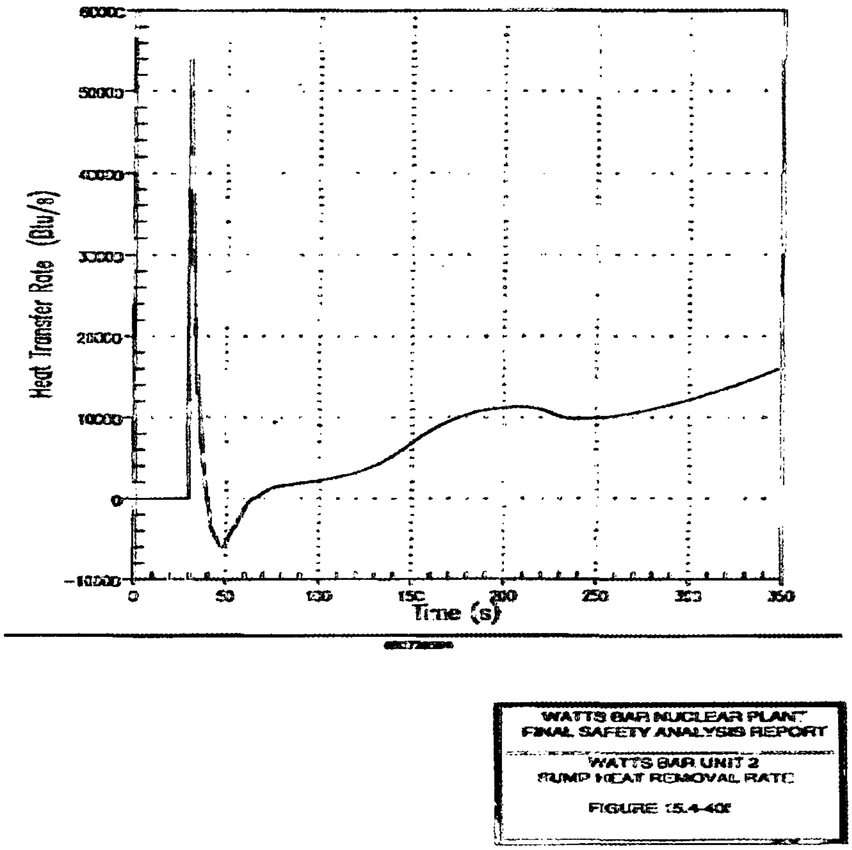


Figure 15.4-40f Watts Bar Unit2 Sump Heat Removal Rate

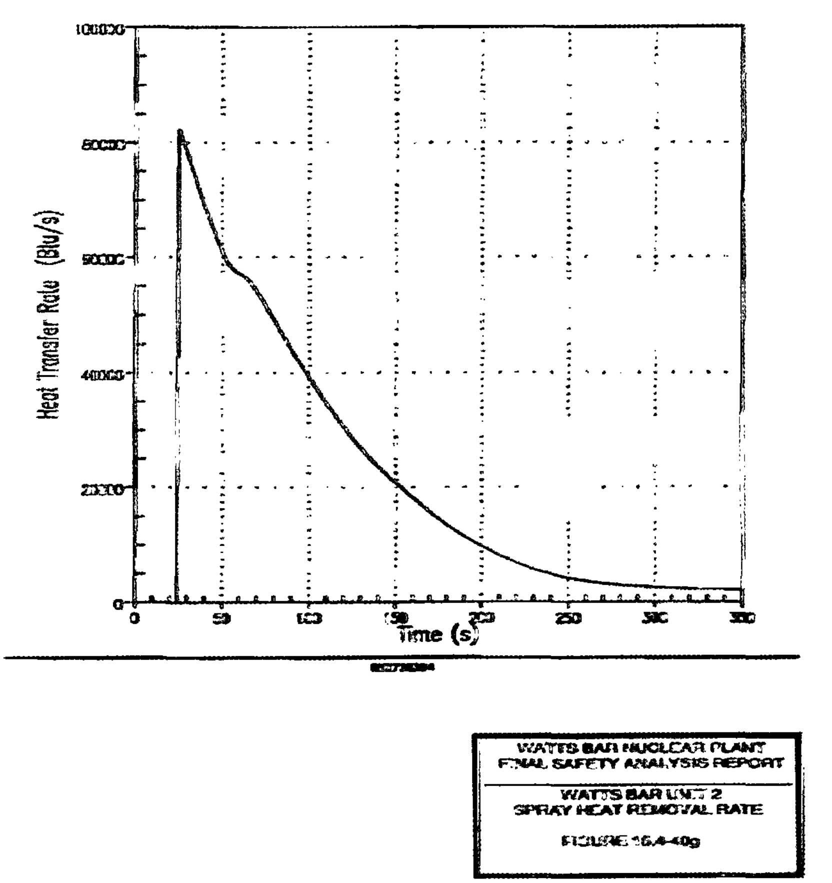
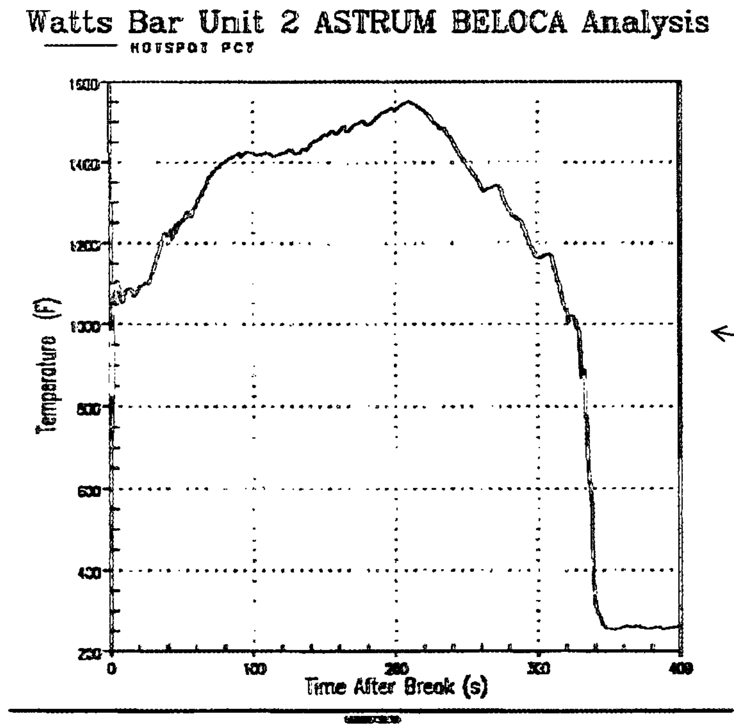


Figure 15.4-40g Watts Bar Unit2 Sump Heat Removal Rate



Update per WBT-D-4396 NP - Attachment Figure 1

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

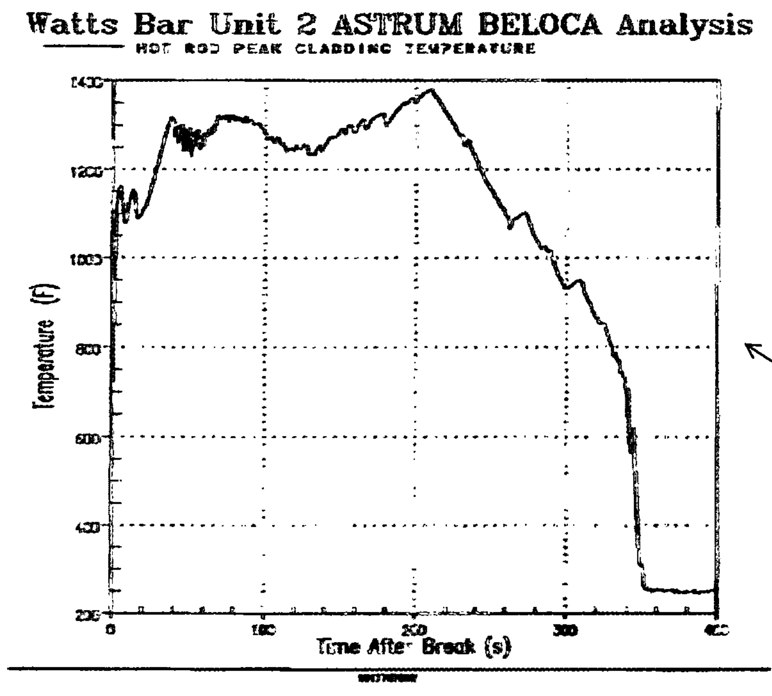
WATTS BAR UNIT 2
LIMITING PCT CASE
HOTSPOT PCT AT THE
LIMITING ELEVATION

FIGURE 15.4-41a

and WC/T PCT

Figure 15.4-41~~a~~ Watts Bar Unit 2 Limiting PCT Case ~~Hotspot~~ PCT At The Limiting Elevation

HOTSPOT



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
LIMITING PCT CASE
WC/T PCT

FIGURE 15.4-41b

delete this figure

~~Figure 15.4-41b Watts Bar Uni 2 Limiting PCT Case WC/T PCT~~

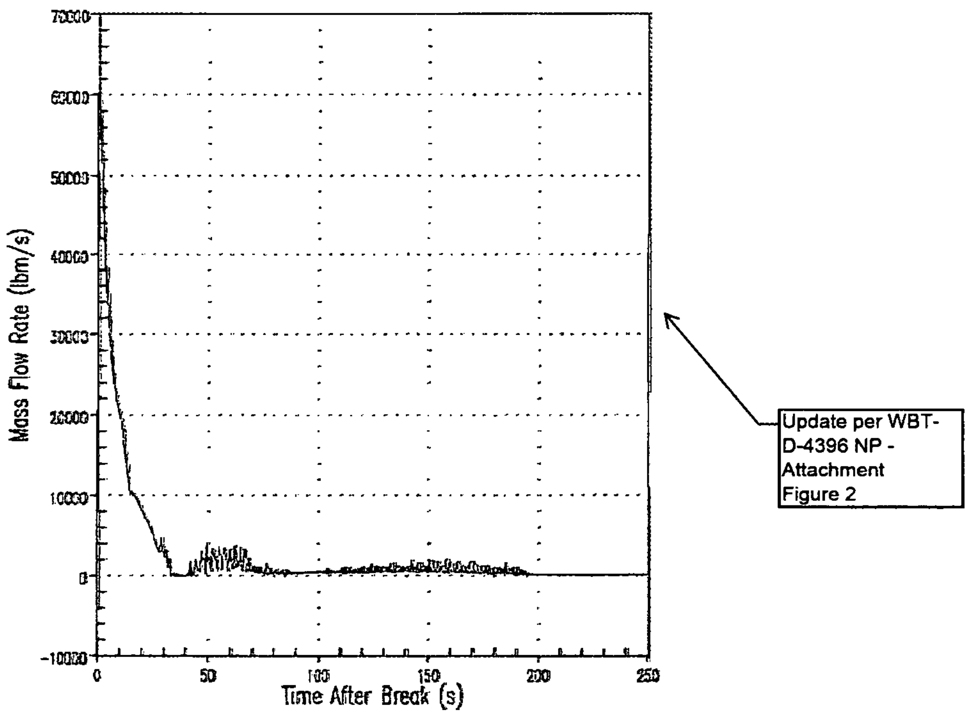
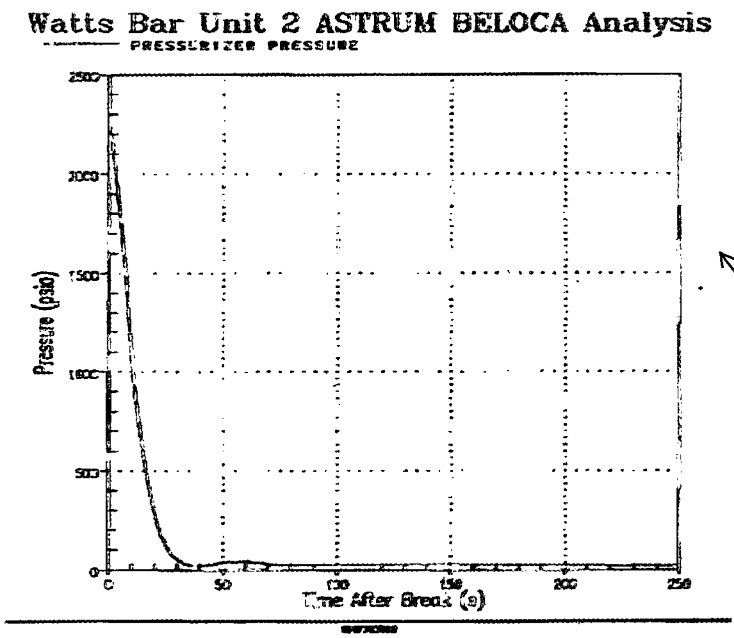


Figure 15.4-42 Watts Bar Unit 2 Limiting PCT Case Break Flow



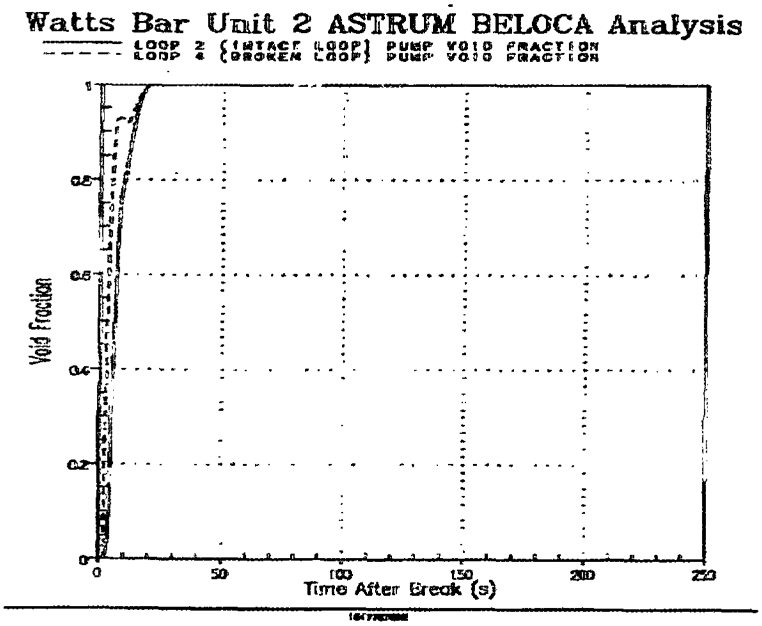
Update per WBT-D-4396 NP - Attachment Figure 3

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
LIMITING PCT CASE
PRESSURIZER PRESSURE

FIGURE 15.4-43

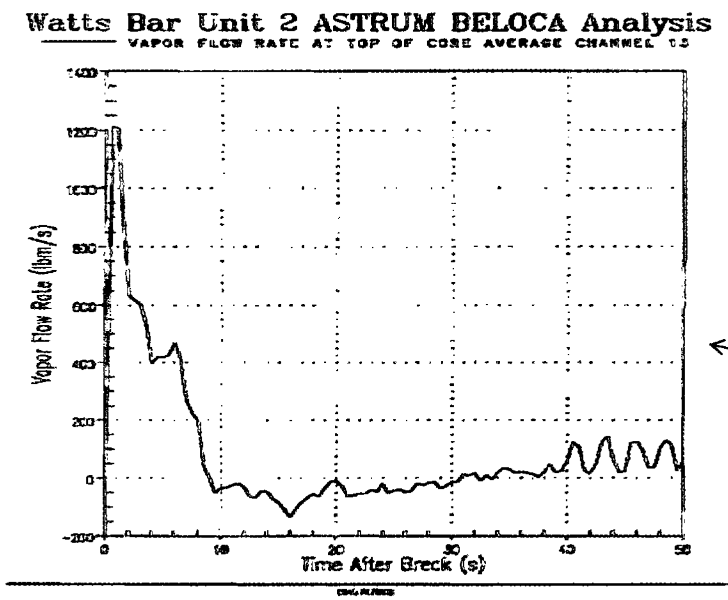
Figure 15.4-43 Watts Bar Unit 2 Limitig PCT Case Pressurizer Pressure



Update per WBT-D-4396 NP - Attachment Figure 4

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
WATTS BAR UNIT 2
LIMITING PCT CASE BROKEN AND INTACT LOOP VOID FRACTION
FIGURE 15.4-44

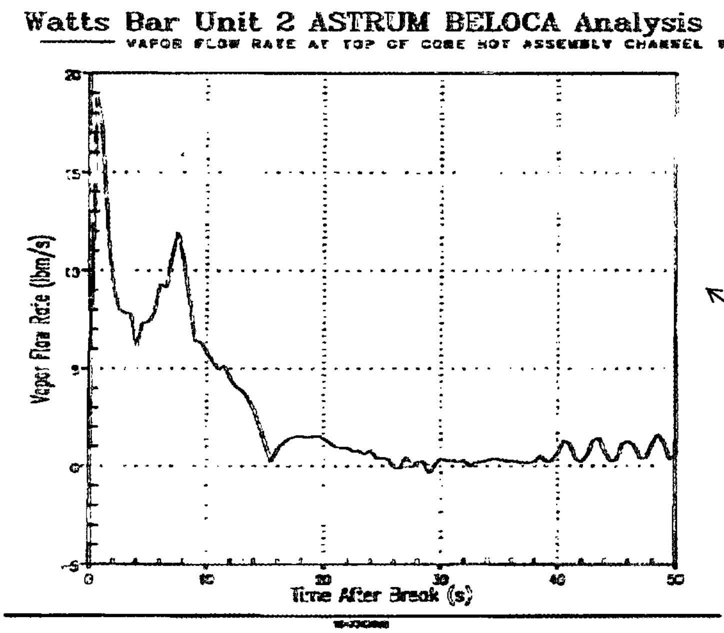
Figure 15.4-44 Watts Bar Unit 2 Limiting PCT Case Broken And Intact Loop Void Fraction



Update per WBT-D-4396 NP - Attachment Figure 5

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
WATTS BAR UNIT 2
LIMITING PCT CASE VAPOR FLOW AT TOP OF CORE AVERAGE CHANNEL
FIGURE 15.4-45

Figure 15.4-45 Watts Bar Unit 2 Limiting PCT Case Vapor Flow At Top Of Core Average Channel



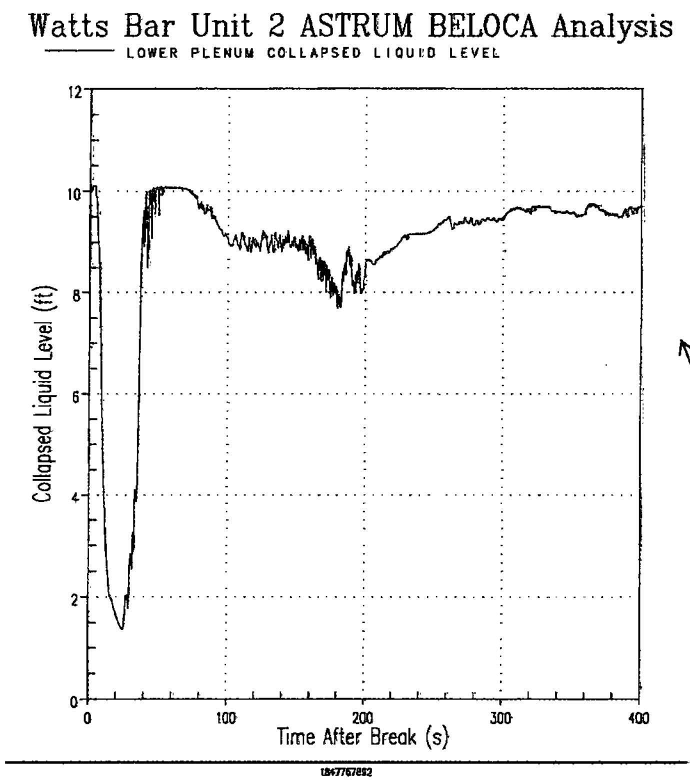
Update per WBT-D-4396 NP - Attachment Figure 6

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
LIMITING PCT CASE VAPOR FLOW AT TOP OF HOT ASSEMBLY CHANNEL

FIGURE 15.4-46

Figure 15.4-46 Watts Bar Unit 2 Limiting PCT Case Vapor Flow At Top Of Hot Assembly Channel



Update per WBT-D-4396 NP - Attachment Figure 7

Figure 15.4-47 Watts Bar Unit 2 Limiting PCT Case Lower Plenum Collapsed Liquid Level

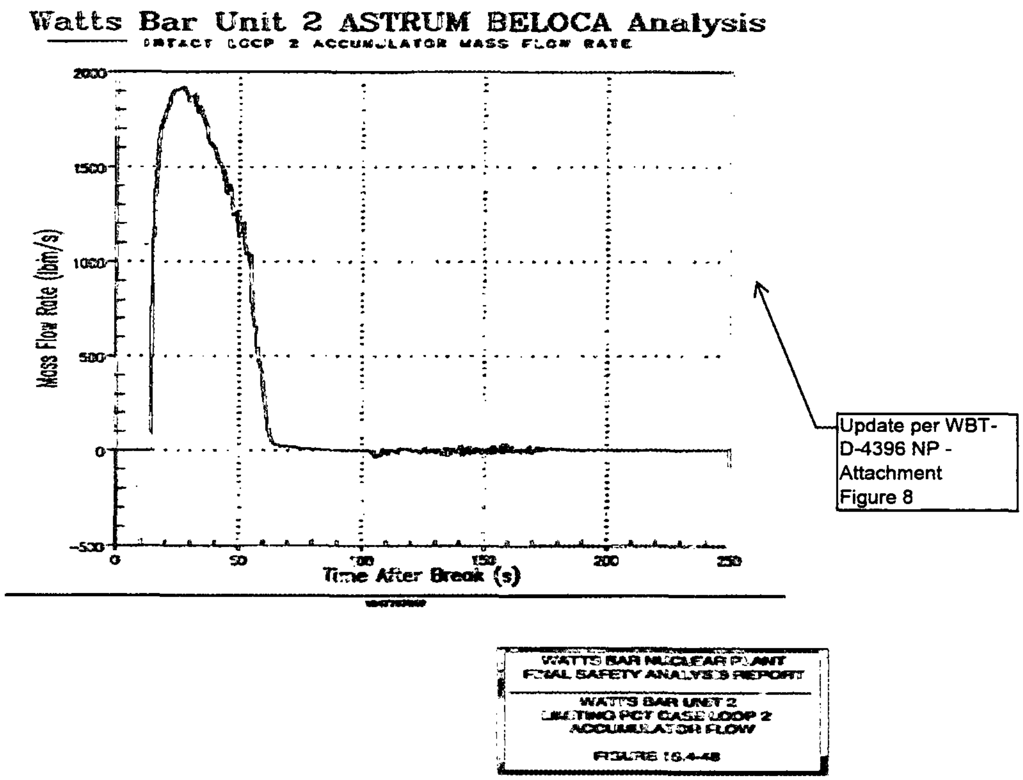
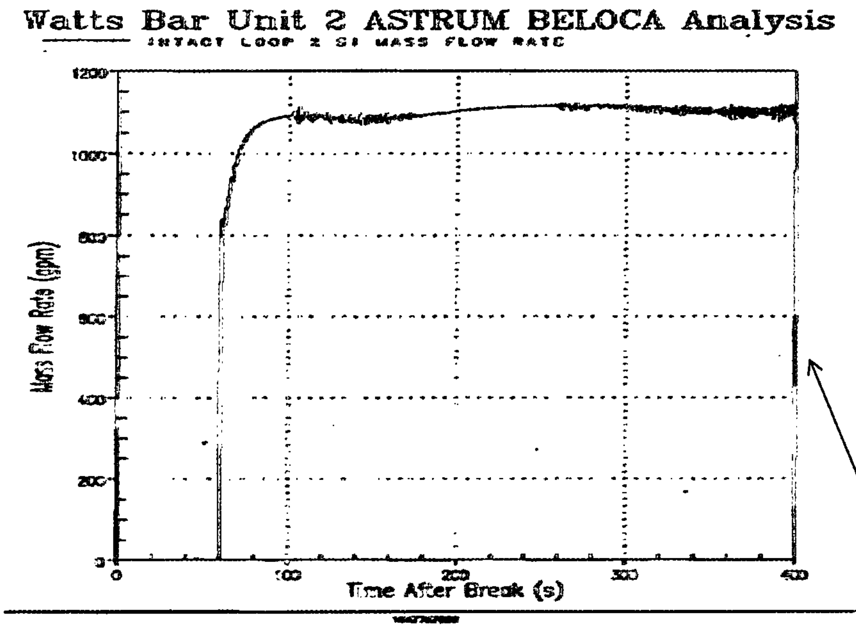


Figure 15.4-48 Watts Bar Unit 2 Limiting PCT Case Loop 2 Accumulator Flow



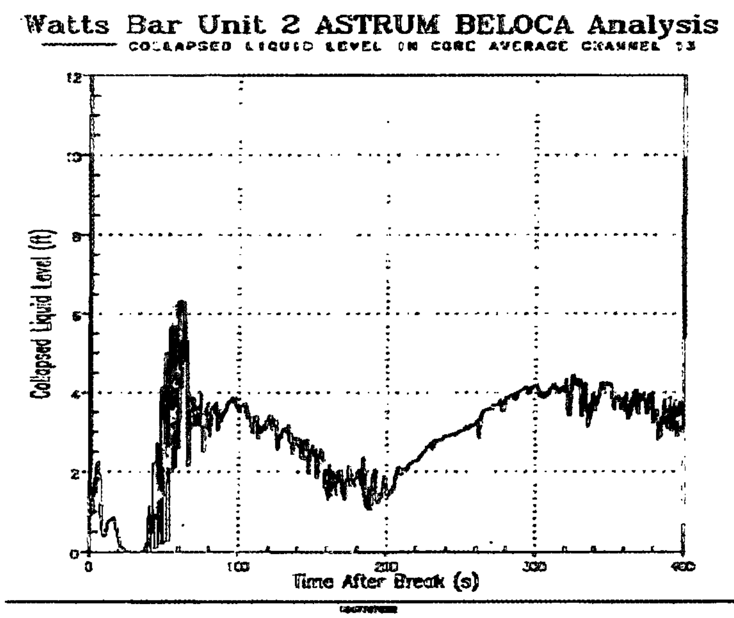
Update per WBT-D-4396 NP - Attachment Figure 9

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
LIMITING PCT CASE LOOP 2
SAFETY INJECTION FLOW

FIGURE 15.4-49

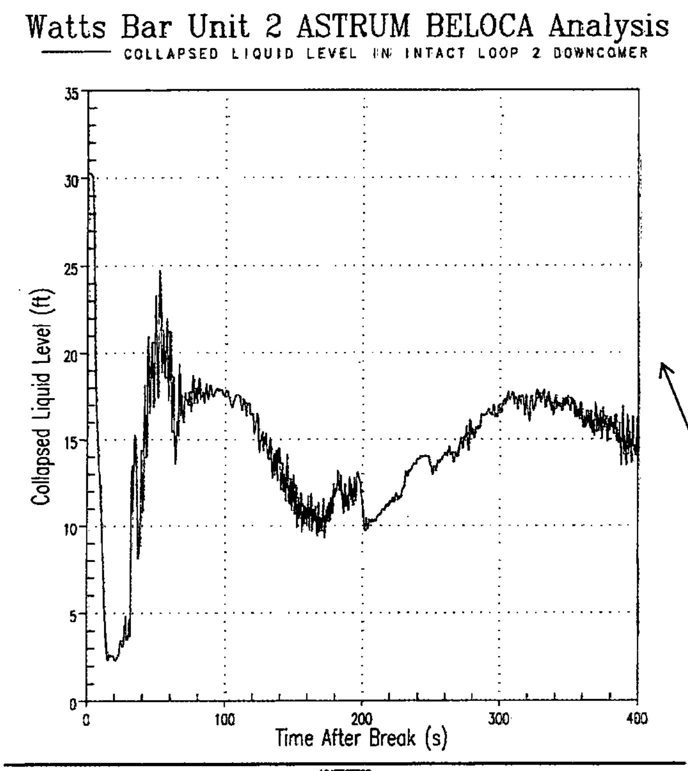
Figure 15.4-49 Watts Bar Unit 2 Limiting PCT Case Loop 2 Safety Injection Flow



Update per WBT-D-4396 NP - Attachment Figure 10

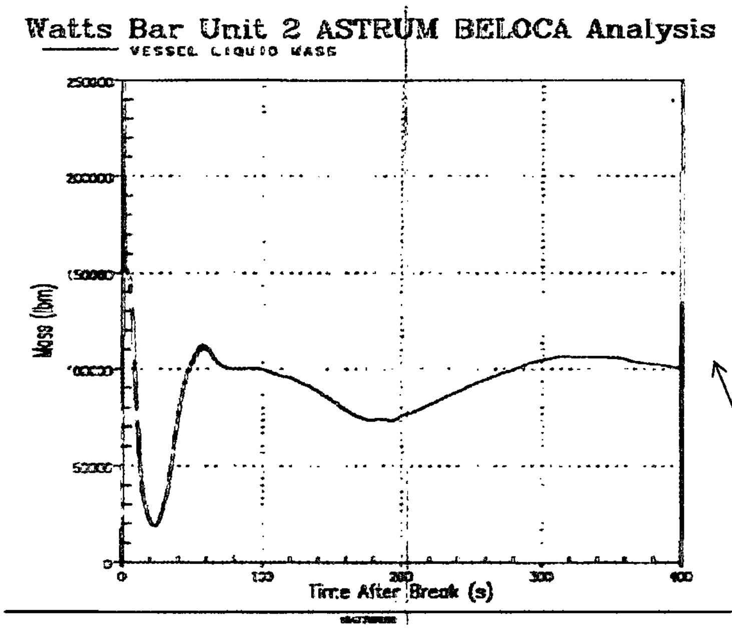
WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
WATTS BAR UNIT 2
LIMITING PCT CASE CORE AVERAGE CHANNEL COLLAPSED LIQUID LEVEL
FIGURE 15.4-50

Figure 15.4-50 Watts Bar Unit 2 Limiting PCT Case Core Average Channel Collapsed Liquid Level



Update per WBT-D-4396 NP - Attachment Figure 11

Figure 15.4-51 Watts Bar Unit 2 Limiting PCT Case Loop 2 Downcomer Collapsed Liquid Level



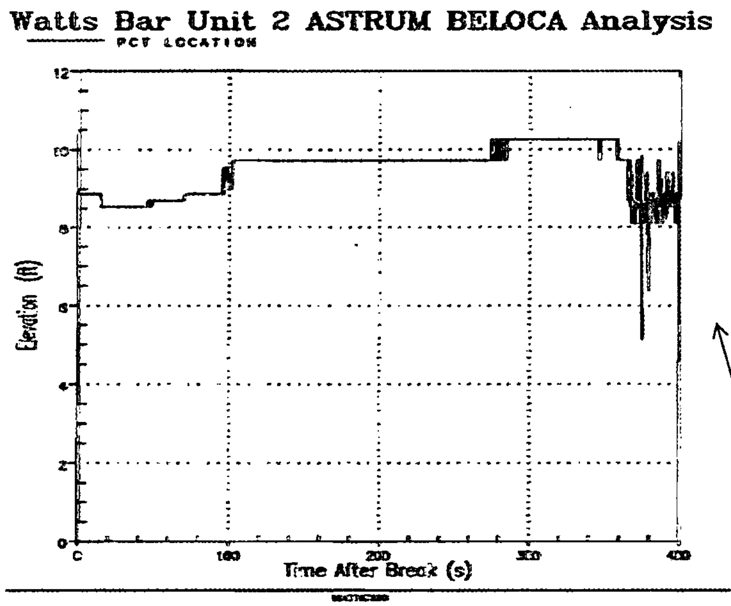
Update per WBT-D-4396 NP - Attachment Figure 12

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
LIMITING PCT CASE
VESSEL FLUID MASS

FIGURE 15.4-52

Figure 15.4-52 Watts Bar Unit 2 Limiting PCT Vessel Fluid Mass



Update per WBT-D-4396 NP - Attachment Figure 13

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
LIMITING PCT CASE
PCT LOCATION

FIGURE 15.4-53

Figure 15.4-53 Watts Bar Unit 2 Limiting PCT Case PCT Location

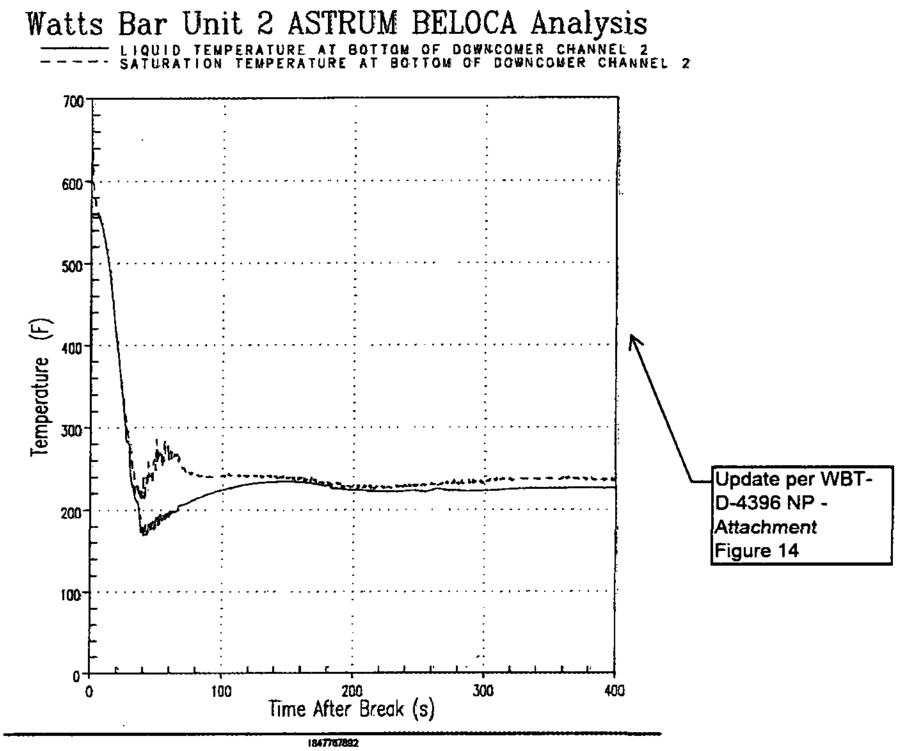
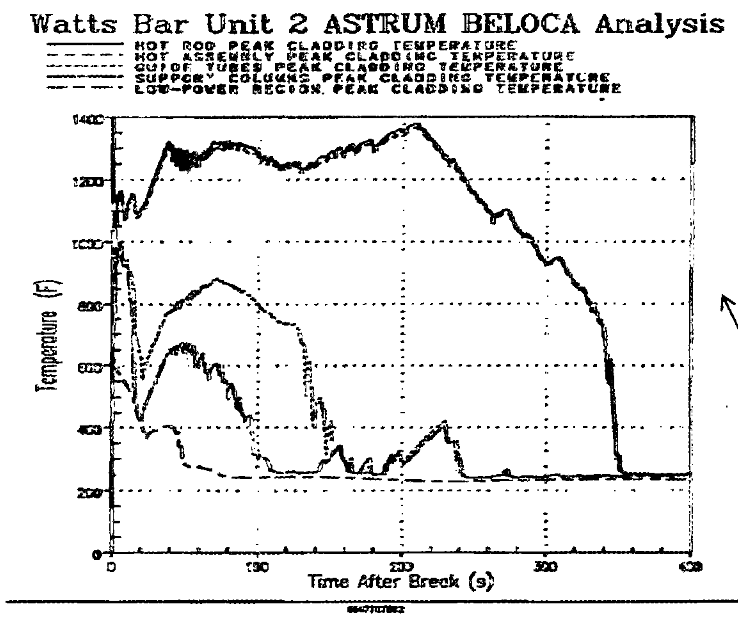


Figure 15.4-54 Watts Bar Unit 2 Limiting PCT Case Liquid And Saturation Temperature At Bottom Of Downcomer



Update per WBT-D-4396 NP - Attachment Figure 15

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
LIMITING PCT CASE
PCT FOR ALL RODS

FIGURE 15.4-55

Figure 15.4-55 Watts Bar Unit 2 Limiting PCT Case PCT For All Rods

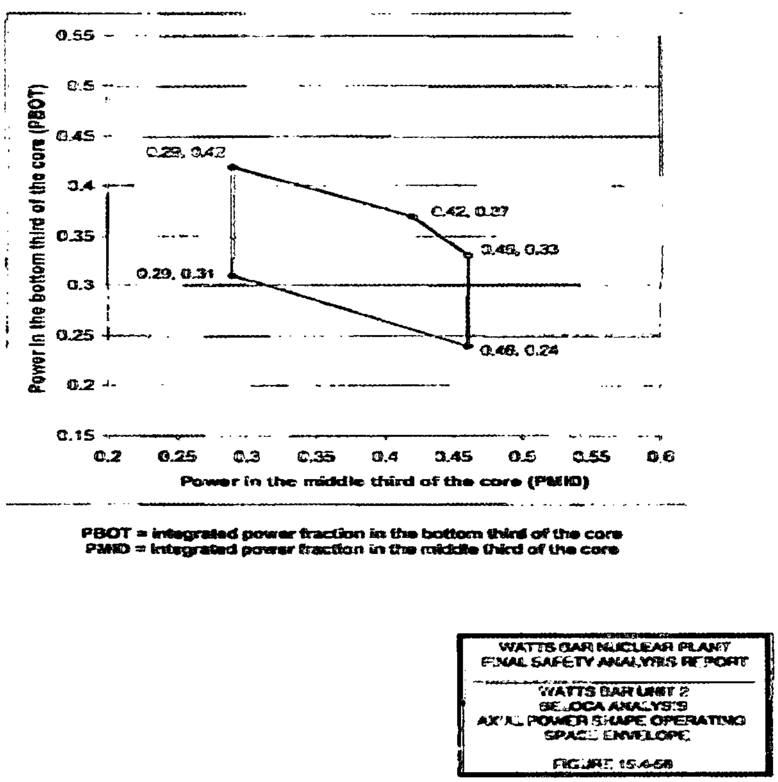


Figure 15.4-56 Watts Bar Unit 2 BELOCA Analysis Axial Power Shape Operating Space Envelope

FSAR INSERTS
(2 pages follow)

Insert A (to page 15.4-4)

The Watts Bar 2 ASTRUM LBLOCA uses a plant-specific adaptation of the ASTRUM methodology that includes explicit modeling of fuel thermal conductivity degradation (TCD), as well as a larger sampling range for rod internal pressure (RIP) uncertainty.

Insert B (to page 15.4-4)

WCAP-16009-P-A [49] states that the ASTRUM methodology is based on the frozen code version WCOBRA/TRAC MOD7A, Revision 6. WCOBRA/TRAC MOD7A, Revision 8-T2 was used for the execution of ASTRUM Uncertainty Studies for Watts Bar Unit 2. The confirmatory analysis (paragraph "2) Determination of Plant Operating Conditions") were executed with WCOBRA/TRAC MOD7A Revision 7.

The Nuclear Regulatory Commission (NRC) approved Westinghouse Best-Estimate Loss-of-Coolant Accident (BELOCA) ASTRUM methodology [49] is based on the PAD 4.0 fuel performance code [51]. PAD 4.0 was licensed without explicitly considering fuel thermal conductivity degradation (TCD) with burnup. Explicit modeling of TCD in the fuel performance code leads directly to increased fuel temperatures (pellet radial average temperature) as well as other fuel performance related effects beyond beginning-of-life. Since PAD provides input to the large-break LOCA analysis, this will tend to increase the stored energy at the beginning of the simulated large-break LOCA event. This in turn leads to an increase in Peak Cladding Temperature (PCT) if there is no provision to credit off-setting effects. In addition, a different fuel thermal conductivity model in WCOBRA/TRAC and HOTSPOT was used to more accurately model the fuel temperature profile when accounting for TCD.

In order to mitigate the impact of the increasing effect of pellet TCD with burnup, the large-break LOCA evaluation of second/third Cycle fuel utilized reduced peaking factors from those shown directly in FSAR Table 15.4-19. The reduced peaking factors are limited to the following application: Burndown credit for the hot rod and hot assembly is taken for higher burnup fuel in the second/third cycle of operation. The Watts Bar Unit 2 peaking factor values utilized in this analysis are shown in Table 15.4-24. Note that the beginning to middle of life values are retained at their direct Table 15.4-19 values.

It should be noted that evaluation of fuel in its second/third cycle of irradiation is beyond the first cycle considered in the approved ASTRUM Evaluation Model (EM), but was considered in the analysis when explicitly modeling TCD to demonstrate that conformance to the acceptance criteria is met for the second/third cycle fuel.

In addition to the standard uncertainty calculations, the Watts Bar 2 LBLOCA analysis sampled a larger rod internal pressure (RIP) uncertainty than originally included in the ASTRUM methodology [49]. It was discovered that the as-approved sampling range did not bound the plant-specific rod internal pressure uncertainties for Watts Bar 2. Therefore, the approved sampling range was expanded to bound the Watts Bar 2 plant-specific data.

Insert C (to page 15.4-7)

The confirmatory configuration analysis was performed previous to the ASTRUM uncertainty calculations prior to the identification of the TCD issue and associated PAD data. However, as no miscellaneous plant configuration changes were introduced, and the effects of TCD are minimal for the confirmatory analysis, the limiting plant configuration (Referred to as the Reference Transient) was judged to remain the same.

Insert D (to page 15.4-7)

The Table 15.4-16 mass and energy releases are taken from the 'Reference Transient' case of Section 15.4.1.1.2, which did not include the fuel TCD modeling. The conservatively low containment backpressure from this LOTIC-2 study is bounding since the core stored energy increases when explicitly modeling fuel TCD, which would tend to increase energy released through the break and hence increase the containment pressure.

Insert E (to page 15.4-8)

PCT and MLO/CWO transients are double ended cold leg guillotine breaks with an effective break area of 1.911, and 2.0968 respectively (note that the limiting MLO and CWO arise from the same case),

Insert F (to page 15.4-10)

An evaluation of IFBA fuel including the effects of pellet TCD was performed, and shows that IFBA fuel is limiting for MLO but not for PCT. The AOR PCT and MLO results in Tables 15.4-18a and 15.4-18b reflect the higher results of IFBA/non-IFBA.

Insert G (to page 15.4-47)

(51) "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1 with Errata (Proprietary), July 2000.

Table Insert 1(follows Table 15.4-23)

Add Table 15.4-24 as follows:

Table 15.4-24 Summary of Peaking Factor Burndown Analyzed by the Best-Estimate Large-Break LOCA Analysis for Watts Bar Unit 2

Hot Rod Burnup (MWD/MTU)	FdH (with uncertainties)	FQ Transient (with uncertainties)	FQ Steady-state (without uncertainties)
0	1.65 (1)	2.50 (1)	2.00 (1)
30000	1.65 (1)	2.50 (1)	2.00 (1)
60000	1.525	2.25	1.800
62000	1.525	2.25	1.800
Note 1: Same Value as Table 15.4-19 (Note FQ SS is titled 'SS depletion' therein)			

ENCLOSURE 5
Tennessee Valley Authority
Watts Bar Nuclear Plant, Unit 2
Docket No. 50-391

NEW COMMITMENT

The information provided in Enclosure 4 will be incorporated into the WBN Unit 2 FSAR by Amendment 110.