Section C

Westinghouse Electric Company
Nuclear Services
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: greshaja@westinghouse.com

LTR-NRC-12-46

June 13, 2012

Subject: Supplemental Information to WCAP-17524, "AP1000 Core Reference Report" to Address Thermal Conductivity Degradation (Proprietary/Non-Proprietary)

References: 1) LTR-NRC-12-32, "Westinghouse Transmittal of Schedule to Address Thermal Conductivity Degradation (TCD) for the AP1000 Core Reference Report (Project #0793)" dated April 5, 2012.

2) WCAP-17524-P, Revision 0, "AP1000 Core Reference Report," March 2012.

Attached are the proprietary and non-proprietary versions of, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to Address Thermal Conductivity Degradation." As communicated in the Westinghouse transmittal letter, LTR-NRC-12-32 (Reference 1), Westinghouse is herein providing the impacts of thermal conductivity degradation (TCD) on the Large Break Loss of Coolant Accident (LBLOCA) peak clad temperature (PCT) with respect to the Advanced First Core as described in the Core Reference Report (Reference 2) for the AP1000®[1] plant.

Also enclosed is:

1. One (1) copy of the Application for Withholding Proprietary Information from Public Disclosure, AW-12-3494 (Non-Proprietary), with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

---

1 AP1000 is a registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference AW-12-3494, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance

Enclosures

Westinghouse

Westinghouse Electric Company
Nuclear Services
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: greshaja@westinghouse.com

AW-12-3494

June 13, 2012

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-12-46 P-Attachment, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to Address Thermal Conductivity Degradation" (Proprietary)

Reference: Letter from J. A. Gresham to Document Control Desk, LTR-NRC-12-46, dated June 13, 2012

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-12-3494 accompanies this Application for Withholding Proprietary Information from Public Disclosure, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject 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 proprietary aspects of the application for withholding or the accompanying affidavit should reference AW-12-3494, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared J. A. Gresham, 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:

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 13th day of June 2012

Cynthia Olesky

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Cynthia Olesky, Notary Public
Manor Boro, Westmoreland County
My Commission Expires July 16, 2014
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance, in Nuclear Services, 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 LTR-NRC-12-46 P-Attachment, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to Address Thermal Conductivity Degradation" (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter, LTR-NRC-12-46, and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with Westinghouse's request for NRC approval of WCAP-17524, and may be used only for that purpose.

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

(a) Obtain NRC approval of the Advanced First Core for the AP1000 plant, as documented in WCAP-17524-P, "AP1000 Core Reference Report."

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of this information to its customers for the purpose of assisting customers in obtaining license changes for the AP1000 pressurized water reactor (PWR).

(b) This document established a portion of the licensing basis for the AP1000 PWR.

(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 technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

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 for generic and/or plant-specific review and approval.

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

Supplemental Information to WCAP-17524, "AP1000 Core Reference Report" to Address Thermal Conductivity Degradation

June 2012

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

1.0 Background

Thermal Conductivity Degradation (TCD) is a physical phenomenon in which the material properties of the fuel (pellets) are affected over the course of in-pile operation (burnup) resulting in a reduced ability to transfer energy from the pellet to the coolant. Consequently, stored energy in the pellets will be higher at burnup when TCD is considered than when TCD is not considered (other effects, such as power fall-off with burnup must be considered to account for the overall impact of TCD). Current Westinghouse Evaluation Models for fuel performance and safety analysis do not explicitly model this effect, although these models do contain conservatism which offset the effects of TCD based on comparisons with test data. These trade-offs are known and documented, and provide an acceptable basis to conclude that TCD does not represent a safety issue. However, the NRC has recently indicated their concern that these effects may pose a compliance question for which they believe the application of generic Evaluation Model conservatism may not be appropriate.

After the submittal of the Core Reference Report Westinghouse transmitted LTR-NRC-12-32, "Westinghouse Transmittal of Schedule to Address Thermal Conductivity Degradation (TCD) for the AP1000 Core Reference Report" (Reference 1). In this transmittal Westinghouse committed to providing an estimate of the effect of TCD on the Peak Clad Temperature (PCT) calculated for the Large Break Loss of Coolant Accident (LBLOCA) analysis presented in WCAP-17524, "AP1000 Core Reference Report" (Reference 2).

The supplemental information provided herein includes a description of the methods used to evaluate the impact of TCD on the LBLOCA analysis presented in the Core Reference Report and an estimate of the impact on the reported LBLOCA PCT results presented in Appendix F of the Core Reference Report.

2.0 Description of the Issue

Fuel pellet TCD and peaking factor burndown were not explicitly considered in the AP1000®[1] plant LBLOCA analysis presented in the Core Reference Report (Reference 2). NRC Information Notice 2011-21 (Reference 3) notified addressees of recent information obtained concerning the impact of irradiation on fuel thermal conductivity and its potential to cause significantly higher predicted PCT results in realistic emergency core cooling system (ECCS) evaluation models. This evaluation provides an estimated effect of TCD on PCT for the Emergency Core Cooling System (ECCS) in the AP1000 plant design.

Fuel performance data that accounts for fuel pellet TCD (using an unlicensed model) was used as input to the AP1000 plant evaluation. The new PAD fuel performance data was generated with a representative model that includes explicit modeling of fuel pellet TCD. Therefore the evaluations performed consider the fuel pellet TCD effects cited in NRC Information Notice 2011-21 (Reference 3).

---

1 AP1000 is a registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. Other names may be trademarks of their respective owners.

3.0 Fuel Performance Methodology and Assumptions for Thermal Conductivity Degradation Evaluation

The Nuclear Regulatory Commission (NRC) approved Performance Analysis and Design (PAD) 4.0 code, with NRC-approved models (Reference 4) for in-reactor behavior, is used as the basis for the assessment of the impact of TCD on fuel performance inputs to safety analyses. PAD 4.0 is a best-estimate fuel rod performance model, with established uncertainties for each of the major fuel and cladding performance models.

The licensed PAD 4.0 fuel performance models do not explicitly address the impact of TCD. The fuel thermal conductivity model in PAD 4.0 is:

a,c

[ ]

[

]a,c

a,c

[ ]

[ ]a,c

a,c

[ ]

[

]a,c

[

]a,c

a,c

[

]

Figure 1: Comparison of Measured Minus Predicted Fuel Temperature as a Function of Burnup for PAD 4.0 and PAD 4.0 TCD

4.0 Large Break Loss of Coolant Accident Methodology for Thermal Conductivity Degradation Evaluation

Westinghouse currently employs the ASTRUM best estimate Evaluation Model (EM) methodology for analysis of the AP1000 pressurized water reactor (PWR) large break loss-of-coolant accident:

- 2004 Westinghouse Realistic LBLOCA Evaluation Model Using ASTRUM (Automated Statistical Treatment of Uncertainty Method) (ASTRUM EM, Reference 6)

The ASTRUM EM is executed assuming a LBLOCA to be [

]^{a,c} The basis for the modeling approach and supporting sensitivity studies are discussed in Section 11-2-2 of Reference 6.

The ASTRUM EM was licensed using PAD 4.0. PAD 4.0 fuel temperature calculations indicate that [

]^{a,c}

The ASTRUM EMs uses WCOBRA/TRAC and HOTSPOT for calculation of the thermal-hydraulic and PCT response to a LBLOCA. The WCOBRA/TRAC and HOTSPOT versions used in ASTRUM analysis include options to use a fuel thermal conductivity model that accounts for TCD; these options were not used in the ASTRUM analysis. HOTSPOT also includes the ability to use pellet radial power profiles from WCOBRA/TRAC which are appropriate to the burnup modeled for a given rod. The ability for HOTSPOT to use pellet radial power profiles from WCOBRA/TRAC which are appropriate to the burnup modeled for a given rod was previously reported to the NRC per Reference 7 as a discretionary change; this feature was used in ASTRUM analysis.

Calculations for TCD evaluations will use code versions with these thermal conductivity and pellet radial profile features in order to appropriately initialize the WCOBRA/TRAC and HOTSPOT fuel rod to the input fuel temperatures and pressures from the fuel performance code and determine the impact of TCD with peaking factor burndown on PCT.

[

]^{a,c}

a,c

[ ]

Where:

a,c

[

]

[

]a,c

a,c

[

]

Where:

a,c

[

]

[

]a,c

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. As inferred from the decrease in fuel temperatures and stored energy in Figures 3 and 4 of Reference 9, TCD and peaking factor burndown are inter-related and should be coupled for the purposes of the evaluation. Therefore, the effect of TCD including peaking factor burndown is estimated to be the difference between a compliance PCT and a margin PCT.

The evaluation is based on running [

]a,c

5.0 Large Break Loss of Coolant Accident Peak Clad Temperature Estimate

5.1 Input Parameters and Assumptions

Determining an estimated PCT effect due to TCD and peaking factor burndown at higher PCTs may result in an exaggerated estimated PCT effect because of a calculated run-away zirconium–water reaction which could occur if the analysis contains excessive conservatism. Therefore, in order to evaluate the estimated effect of TCD, conservatisms in the ASTRUM analysis presented in the Core Reference Report (Reference 2) were evaluated. Specifically, the following analysis input change in the LBLOCA analysis presented in the Core Reference Report was evaluated in order to more accurately estimate the impact of TCD:

- Reduction in the as-analyzed F_Q to a value closer to the desired F_Q as defined by the ASTRUM evaluation method for the top two most limiting PCT cases from analysis presented in Reference 2 (see Table 5-1). This reduction removed analysis conservatism associated with using values of F_Q in code executions that significantly exceeded target values.

The evaluation of fuel TCD and peaking factor burndown considered the following additional input parameter changes to the LBLOCA analysis:

- Fuel rod design data with PAD 4.0 + TCD
- Peaking factor burndown shown in Table 5-2

Westinghouse Electric Company utilizes processes which ensure that the LOCA analysis input values conservatively bound the as-designed plant values for those parameters.

5.2 Evaluation Basis

The evaluation method discussed in Section 3.0 and Section 4.0 was used to determine the estimated effect of fuel pellet TCD and peaking factor burndown. First, the integrated PCT was calculated to demonstrate compliance with the 10 CFR 50.46(b)(1) criterion when the analysis input changes and TCD and burndown were considered. Then, the margin PCT was calculated, including only the analysis input changes.

For the integrated PCT calculation, a total of 31 WCOBRA/TRAC executions were performed. The uncertainty attributes of these executions were taken from among the most limiting cases from the original 124-run ASTRUM analysis discussed in the Core Reference Report (Reference 2). The evaluation considered an adequate range of burnup such that the effects of TCD and related burnup effects were captured. HOTSPOT executions were performed for each WCOBRA/TRAC case to consider the effect of local uncertainties for both IFBA (Integral Fuel Burnable Absorber) and non-IFBA fuel.

For the margin PCT calculation, WCOBRA/TRAC executions were performed for the top two PCT cases from the Reference 6 ASTRUM analysis with the reduced conservatism in F_Q described in Table 5-1. The margin PCT result was then determined as the limiting PCT from these two margin cases and the PCT results of the rank 3 through 124 cases from the Reference 6 ASTRUM analysis. The margin PCT

calculation does not include effects of peaking factor burndown, consistent with the as-approved ASTRUM evaluation model (Reference 6).

The estimated effect of TCD was then taken as the difference between the integrated PCT and the margin PCT for the Core Reference Report analysis results.

The same WCOBRA/TRAC and HOTSPOT code versions used in the analysis presented in the Core Reference Report (Reference 2) were used in the calculations for evaluation of TCD and analysis input conservatism.

5.3 Impact on the Large Break Loss of Coolant Accident Analysis Presented in the Core Reference Report

The quantitative evaluation as described above was performed to assess the PCT effect of TCD and peaking factor burndown with other considerations of burnup on the LBLOCA analysis presented in the Core Reference Report (Reference 2). The results of the evaluation have been summarized in Table 5-3. As can be seen in the table there is sufficient margin to address TCD impacts on PCT.

Consistent with the ASTRUM methodology, the most limiting PCT from each evaluation was taken as the representative PCT impact. The limiting integrated PCT case, considering all analysis input changes and TCD and burndown, was 1936°F, less than the 2200°F acceptance criterion.

6.0 Conclusions

With the inclusion of an adjusted degree of conservatism in the F_Q and inclusion of the effects of TCD continued compliance with the requirement of 2200°F of 10 CFR 50.46(b)(1) can be demonstrated. Westinghouse is requesting approval of the revised PCT value as part of the review of WCAP-17524, "AP1000 Core Reference Report." The impact of TCD will be reflected in the Chapter 15 analysis as part of the creation of the approved version.

7.0 References

1.0 LTR-NRC-12-32, "Westinghouse Transmittal of Schedule to Address Thermal Conductivity Degradation (TCD) for the AP1000 Core Reference Report (Project #0793)," April 5, 2012.

2.0 WCAP-17524, "AP1000 Core Reference Report" March 2012.

3.0 NRC Information Notice 2011-21, McGinty, T.J., and Dudes, L. A., "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting From Nuclear Fuel Thermal Conductivity Degradation," December 13, 2011. (NRC ADAMS #ML 113430785).

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

5.0 WCAP-15836-P-A (Proprietary), Harris, W. R., et al., "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," April 2006.

6.0 WCAP-16009-P-A (Proprietary), Frepoli, C., et al., "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

7.0 LTR-NRC-07-23, Maurer, B. F., "U. S. Nuclear Regulatory Commission,10 CFR 50.46 Annual Notification and Reporting for 2006," May 15, 2007.

8.0 NUREG/CR-6534, Volume 4, Lanning, D.D., et al., "FRAPCON-3 Updates, Including Mixed-Oxide Fuel Properties," May 2005.

9.0 McGinty, T. J. (NRC) to Gresham, J. A. (Westinghouse), "Nuclear Fuel Thermal Conductivity Degradation Evaluation for Light Water Reactors Using Westinghouse Codes and Methods (TAC NO. ME5186)," December 16, 2011.

Table 5-1: Reduced Conservatism in Reference 2 Analysis FQ Considered in the Evaluation of TCD

Case	FQ Conservatism[(1),(2)] [%]	Adjusted FQ Conservatism[(1),(3)] [%]
A	15	0.4
B	12	0.2

(1) Numbers reflect the percentage by which the as-analyzed FQ in the run exceeded the desired FQ as defined by the ASTRUM evaluation method.
(2) FQ conservatism in runs executed for Reference 5 ASTRUM analysis.
(3) FQ conservatism in the runs executed for evaluation of analysis input changes, and in runs executed for evaluation of fuel TCD and peaking factor burndown with the same analysis input changes.

Table 5-2: Peaking Factors Assumed in the Evaluation of TCD

Rod Burnup (MWD/MTU)	FDH[(1)(2)]	FQ Transient [(1)]	FQ Steady-State
0	1.72	2.60	2.10
30,000	1.72	2.60	2.10
49,000	1.55	2.30	1.85
65,000	1.55	2.30	1.85

(1) Includes uncertainties.
(2) Hot assembly average power follows the same burndown, since it is a function of FdH .

Table 5-3: Summary of PCT Results for Various Evaluations[(1)]

Core Reference Report PCT [°F]	Core Reference Report Margin PCT[(2)] [°F]	Core Reference Report PCT with reduction in input conservatisms and including TCD effects [°F]
1867	1797	1936

(1) All values contain a 2°F bias for sensitivity to passive residual heat removal heat exchanger (PRHR) operation.
(2) Reflects reduction in input conservatism described in Table 5-1.

Section D

A non-proprietary version of this document is not available

Section E

Westinghouse

Westinghouse Electric Company
Nuclear Services
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: greshaja@westinghouse.com

LTR-NRC-12-86

January 2, 2013

Subject: Westinghouse Response to NRC RAIs on WCAP-17524, "AP1000 Core Reference Report" (Proprietary/Non-Proprietary).

Enclosed are copies of the proprietary and non-proprietary versions of the responses to the NRC RAIs for WCAP-17524, "AP1000 Core Reference Report" dated March 2012.

Also enclosed is:

1. One (1) copy of the Application for Withholding Proprietary Information from Public Disclosure, AW-12-3579 (Non-Proprietary), with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference AW-12-3579 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J.A. Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

Westinghouse

Westinghouse Electric Company
Nuclear Services
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: greshaja@westinghouse.com

AW-12-3579

January 2, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-12-86 P-Attachment, "Westinghouse Response to NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report'" (Proprietary)

Reference: Letter from J. A. Gresham to Document Control Desk, LTR-NRC-12-86, dated January 2, 2013

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-12-3579 accompanies this Application for Withholding Proprietary Information from Public Disclosure, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject 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 proprietary aspects of the application for withholding or the accompanying affidavit should reference AW-12-3579 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, 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:

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 2nd day of January 2013

Anne M. Stegman
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, in Nuclear Services, 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 LTR-NRC-12-86 P-Attachment, "Westinghouse Response to NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report'" (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter, LTR-NRC-12-86, and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the review of WCAP-17524, and may be used only for that purpose.

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

(a) Obtain NRC approval of the **AP1000**® Pressurized Water Reactor (PWR) Advanced First Core, as documented in WCAP-17524, "AP1000 Core Reference Report".

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 assisting customers in obtaining license changes for the **AP1000** PWR.

(b) This document establishes a portion of the licensing basis for the **AP1000** PWR.

(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 technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

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 for generic and/or plant-specific review and approval.

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

Westinghouse Response to NRC RAIs on WCAP-17524, "AP1000 Core Reference Report" (Non-Proprietary)

January 2013

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

CRR-002 *(codes and methods)*

The CRR includes various updates to the referenced codes and methodologies used in the nuclear design. WCAP-10965-P-A, "Qualification of the New Pin Power Recovery Methodology," was added to reflect an updated methodology to be used along with the ANC code. The staff's SER (ML102350046) states that this methodology is acceptable as long as the nuclear data generated as input to ANC originates from the PARAGON and NEXUS code systems. During the Phase 2 audit, staff asked if there were any instances where PHOENIX-P is used instead of PARAGON when generating data for ANC since the SER conclusions for the pin power recovery methodology only give approval for use of the methodology with the PARAGON/NEXUS/ANC code system. Westinghouse stated that data from PHOENIX-P is used in a limited capacity with ANC. For clarity, list all PHOENIX-P uses for calculations supporting updates to FSAR Revision 19 for the CRR and justify why the limitations and conditions in the staff SER mentioned above are not violated.

Westinghouse Response to CRR-002

The calculations supporting updates for the CRR were performed with an ANC model that used the PARAGON and NEXUS code systems to generate all cross sections for the active fuel region, where the New Pin Power Recovery Methodology is applied.

PHOENIX-P based cross sections were used only for the non-fuel radial reflector region located outside of the active fuel. This region consists of structural steel and water. No pin power recovery calculations are performed within the reflector region. The purpose of including reflector nodes in the ANC model is to establish the correct neutron leakage at the active fuel boundary. The non-fuel reflector is modeled in ANC only on a nodal level, using homogenized 2-group macroscopic cross sections which are not sensitive to the use of PARAGON or PHOENIX-P. The use of PHOENIX-P based cross sections for the radial reflector is primarily a legacy issue caused by a lack of automation within the PARAGON code system for performing this type of calculation. Westinghouse intends to eventually move all cross section generation for reflector regions to the PARAGON code.

Since only PARAGON/NEXUS based cross sections are used in active fuel region where the New Pin Power Recovery Methodology is employed, the conditions in the staff's SER (ML102350046) are not violated.

It should be noted that the reference to the PHOENIX-P 70-group cross section library discussed in Appendices A and E, in Section 4.3.3.2 of WCAP-17524-P is incorrect in that it specifies that the PHOENIX-P library is derived mainly from ENDF/B-V files. The 70-group PHOENIX-P library currently in use is actually based mainly on ENDF/B-VI files. The old 42-group PHOENIX-P library, which was based on ENDF/B-V files, is no longer in use. Specifically, Westinghouse intends to make the following corrections in the final approved version of WCAP-17524-P.

Section 4.3.3.2 5th Paragraph, 1st Sentence (current):

PHOENIX-P employs a 70 energy group library derived mainly from ENDF/B-V files (Reference 71).

Section 4.3.3.2 5th Paragraph, 1st Sentence (corrected):

PHOENIX-P employs a 70 energy group library derived mainly from ENDF/B-VI files (Reference 71).

Section 4.3.5, Reference 71 (current):

71. Ford, W. E., et. al., "CSRL-V: Processed ENDF/B-V 227-Neutron-Group and Point-wise Cross Section Libraries for Criticality Safety, Reactor and Shielding Studies," NUREF/CR-2306, ORNL/CSD/TM-160 (1982).

Section 4.3.5, Reference 71 (corrected):

71. McLane, V., et. al., "ENDF-201, ENDF/B-VI Summary Documentation," BNL-NCS-17541, 4th Edition [ENDF/B-VI] Supplement 1, National Nuclear Data Center, Brookhaven National Laboratory (1996).

CRR-003 *(codes and methods)*

The CRR includes an update to DCD Section 15.4.8 describing the rod ejection analysis. The updated DCD section now references the rod ejection calculational methodology described in WCAP-15806-P-A, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," which is based on the standalone SPNOVA code. It appears that the codes supporting the 3-D rod ejection calculation have evolved significantly since staff's 2003 approval of WCAP-15806-P-A. Since the CRR rod ejection analyses are now being performed solely with ANC9.4 due to the migration of the SPNOVA solver into ANC9.4, demonstrate that ANC9.4 produces results that are consistent with previous 3-D rod ejection analyses supported by SPNOVA as a standalone code for the **AP1000** design.

Westinghouse Response to CRR-003

Background

Per the Safety Evaluation Report of WCAP-15806-P-A (Reference 1), the licensing basis of the SPNOVA code supporting the Westinghouse 3-D Rod Ejection methodology includes:

- WCAP-12394-A (proprietary), "SPNOVA - A Multidimensional Static and Transient Computer Program for PWR Core Analysis," (Reference 2) and
- NSD-NRC-96-4679, Letter from Liparulo, N.J. (Westinghouse) to Jones, R.C. (NRC), "Process Improvement to the Westinghouse Neutronics Code System." (Reference 3)

Specifically, NSD-NRC-96-4679 identifies the incorporation of the ANC code's NEM solution methodology into the SPNOVA code as a replacement for the less rigorous Green's matrix solution method previously used in the WCAP-12394-A methodology. The multi-dimensional kinetic solution methodology of WCAP-12394-A was not changed with this implementation of the NEM methodology for static core solutions.

Since this time, a further process improvement has been undertaken to unify the two software implementations (i.e., ANC and SPNOVA) into a single piece of software that can be more efficiently maintained. This was first done in ANC version 9.3.0 code, which is based upon the following licensing reports:

- WCAP-10965-P-A, "ANC – A Westinghouse Advanced Nodal Computer Code" (Reference 4)
- NTD-NRC-95-4533, Letter from Liparulo, N.J. (Westinghouse) to Jones, R.C. (NRC), "Notification to the NRC Regarding Improvements to the Nodal Expansion Method Used in the Westinghouse Advanced Nodal Code (ANC)" (Reference 5)
- WCAP-10965-P-A Addendum 1, "ANC – A Westinghouse Advanced Nodal Computer Code - Enhancement to ANC Rod Power Recovery" (Reference 6)
- WCAP-10965-P-A Addendum 2-A, "Qualification of the New Pin Power Recovery Methodology" (Reference 7)
- WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON" (Reference 8)

- WCAP-16045-P-A Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology" (Reference 9)

The updates subsequent to the original approval of WCAP-10965-P-A improve the static core solution methodology for the ANC code but do not impact the multi-dimensional kinetics methodology previously approved in the SPNOVA code. All of these updates culminated in the release of ANC 9.3.0 – the first release of the ANC version 9 series with multi-dimensional kinetics capability. ANC 9.4.0 was subsequently released and regression testing via the applicable procedures confirmed that the multi-dimensional kinetics capability of the code remained valid.

Description of ANC 9.3.0 Software Development and Testing

As mentioned previously, the SPNOVA multi-dimensional space-time kinetics methodology was first incorporated into a single, unified software product with ANC version 9.3.0. This software development was performed under the Westinghouse Quality Management System (QMS). Specific phases of the software development process, as pertain to the implementation of the multi-dimensional kinetics methodology included:

- Development of Software Requirements: These requirements prescribed the implementation of the methodology consistent with the approved methodology in WCAP-12394-A.
- Design/Implementation: It was ensured that the implementation of the methodology in the ANC version 9.3.0 software product was consistent with the requirements in the previous step.
- Software Test Plan: Considering the addition of significant new functionality to the ANC version 9.3.0 software, the test plan prescribed both validation testing (to confirm the proper implementation of software requirements) and qualification testing (to perform benchmarking tests against the existing SPNOVA code).
- Software Test Results: This included confirmation of all validation tests prescribed in the test plan as well as software regression testing again previous version results.
- Qualification of Results vs. SPNOVA: This included a benchmark comparison test between the two software products for a simulated equivalent problem.

To summarize, testing performed for addition of this new functionality to the ANC code included validation that all prescribed software requirements were met and that the results of the software showed good agreement with the multi-dimensional kinetics methodology as implemented in the SPNOVA code.

With respect to some specific testing performed, the following are highlighted as specifically applicable to demonstration of the migration of the SPNOVA functionality into the ANC software:

The software test plan included the performance of a null transient. The kinetics calculations start from an equilibrium condition for the delayed neutrons and evaluate the transient as initiating from that condition. A null transient represents a scenario where there is no change – so the initial equilibrium conditions should be maintained, assuming a proper implementation. A null transient case is an effective means to demonstrate consistency between the static solution and the kinetics solution. During the validation testing, it was demonstrated that all software requirements were adequately met.

To ensure that the results of the kinetics implementation in ANC version 9.3.0 provides results consistent with previous 3D Rod Ejection analyses using the SPNOVA code, a qualification study was performed to benchmark the results of ANC version 9.3.0 to SPNOVA. This study was performed on a typical 3-loop core (i.e., corresponding to that used in the original 3D Rod Ejection topical report - WCAP-15806-P-A, Reference 1), however, the implementation of the space-time kinetics capability is independent of core size or arrangement. Thus, these results are representative of SPNOVA vs. ANC 9.4.0 performance for an **AP1000®** Pressurized Water Reactor (PWR). When the SPNOVA and ANC problems were run with consistent core models, for consistent ejected rod worth, and with a consistent thermal feedback model, the core power transient as generated by the two codes was demonstrated to be nearly identical, as shown in the figure below:

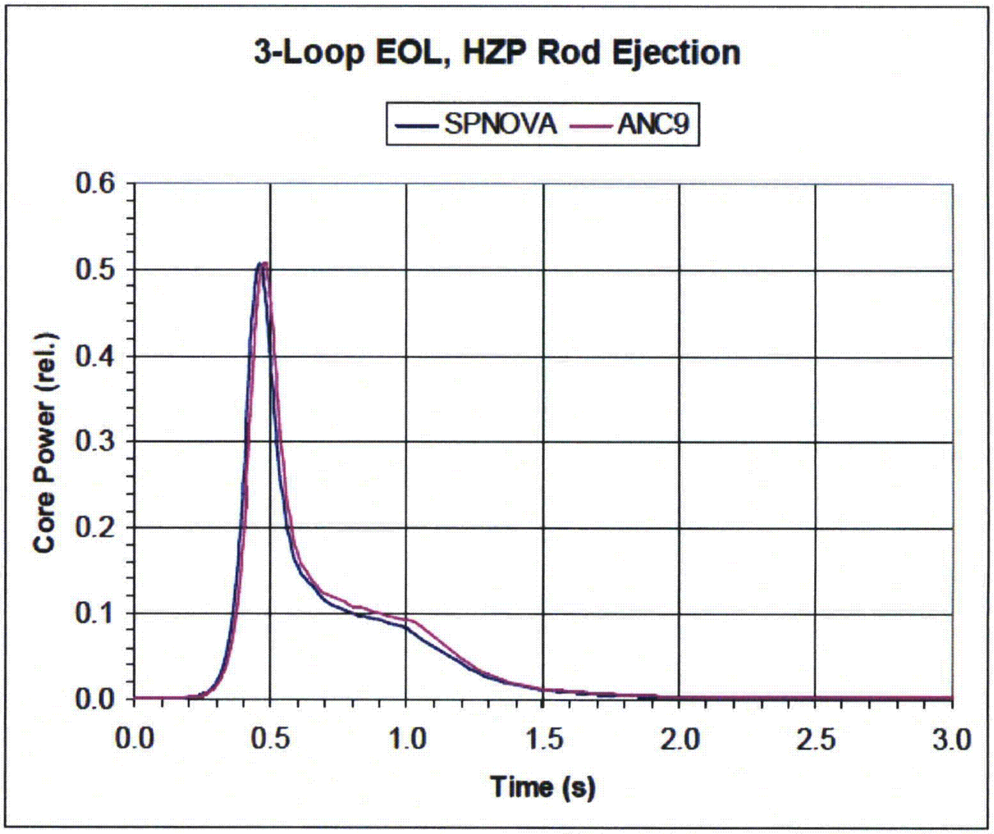

Figure 1: Comparison of equivalent rod ejection problem results between ANC and SPNOVA

Overall, the qualification testing of the multi-dimensional kinetics implementation demonstrated that ANC version 9.3.0 and SPNOVA give equivalent results for a representative rod ejection transient.

References

1. WCAP-15806-P-A, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics"
2. WCAP-12394-A (proprietary), "SPNOVA - A Multidimensional Static and Transient Computer Program for PWR Core Analysis"
3. NSD-NRC-96-4679, Letter from Liparulo, N.J. (Westinghouse) to Jones, R.C. (NRC), "Process Improvement to the Westinghouse Neutronics Code System"
4. WCAP-10965-P-A, "ANC – A Westinghouse Advanced Nodal Code"
5. NTD-NRC-95-4533, Letter from Liparulo, N.J. (Westinghouse) to Jones, R.C. (NRC), "Notification to the NRC Regarding Improvements to the Nodal Expansion Method Used in the Westinghouse Advanced Nodal Code (ANC)"
6. WCAP-10965-P-A Addendum 2-A, "Qualification of the New Pin Power Recovery Methodology"
7. WCAP-10965-P-A Addendum 1, "ANC – A Westinghouse Advanced Nodal Computer Code - Enhancement to ANC Rod Power Recovery
8. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON"
9. WCAP-16045-P-A Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology"

CRR-005 *(LBLOCA Analysis)*

As stated in "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to Address Thermal Conductivity Degradation" (LTR-NRC-12-46 P-Attachment, June 2012), the large-break LOCA analysis, which accounts for the fuel TCD, is performed with a reduction in the as-analyzed total peaking factor F_Q to a value closer to the desired F_Q to remove analysis conservatism. Calculation Note APP-SSAR-GSC-772, Rev. 0, states that this input F_Q conservatism reduction is done by adjusting sampling values of the power integrals at the bottom and middle 1/3 of the core (i.e., PBOT and PMID) to recover the peak linear heat rate margin in the existing analysis.

(a) Describe the process used for adjusting the PBOT and PMID sampled values to obtain F_Q values closer to the desired value.
(b) Explain why this adjusting process is appropriate with respect to the sampling of PBOT/PMID in the ASTRUM methodology.

Westinghouse Response to CRR-005

The **AP1000** plant best estimate large break loss of coolant accident analysis (BELOCA) presented in the Core Reference Report (Reference 1) was performed using the ASTRUM evaluation model (EM) (Reference 2).

In the ASTRUM EM, a method based on non-parametric order statistics is used to combine uncertainty contributors in order to demonstrate a high level of probability that the acceptance criteria specified by 10 CFR 50.46(b)(1), (b)(2) and (b)(3) are not exceeded. This method is described in Reference 2 Section 11. As described in Reference 2, to bound the 95th percentile peak cladding temperature (PCT), maximum local oxidation (MLO), and core-wide oxidation (CWO) values at a 95% confidence level (as a joint-probability singular statement), the top rank (maximum value) of the 124 run-set sample is selected as the estimator for each output parameter. Therefore, for each of 124 coupled WCOBRA/TRAC and HOTSPOT calculations executed as part of an ASTRUM uncertainty analysis, uncertainty contributors are randomly sampled from their respective distributions as input to the calculation; the key uncertainty contributors treated in the ASTRUM EM are summarized in Reference 2 Table 1-7, Table 1-8, Table 1-10 and Table 1-11. Then the bounding estimates of the 95th percentile at 95% confidence (95/95 estimates) are the limiting PCT, MLO and CWO results from the set of 124 cases; they may come from the same case or from two or three different cases.

In the ASTRUM analysis presented in the CRR, the axial power shapes in the top two limiting PCT cases had as-executed maximum total peaking factor FQ values, and therefore peak linear heat rates (PLHRs) higher than the desired values based on the sampled uncertainty parameters for those cases. This was accepted as a discretionary conservatism in the analysis results when the CRR analysis was executed.

Following the submittal of the **AP1000** PWR CRR, additional work was done to quantitatively assess the impact of thermal conductivity degradation (TCD) on the **AP1000** PWR BELOCA PCT results. This work is documented in Reference 3 and is summarized in Reference 4. As part of this work, the conservatism in the PLHR values in the top 2 PCT cases of the CRR analysis was evaluated, which partially offset the effects of TCD. The conservatism was evaluated by executing these cases with revised

axial power shape input. For each case, axial power shapes were generated which had maximum FQ less than 1% higher than the desired value based on the sampled values for those cases. To generate the revised axial power shapes, the integrated power in the bottom one-third of the core (PBOT) and integrated power in the middle one-third of the core (PMID) were shifted from [

]a,c

First, the axial power shape generation process in an ASTRUM analysis is discussed. Next, the process used to adjust the PBOT and PMID sampled values in the top 2 PCT cases from the CRR analysis, to obtain FQ and therefore PLHR values closer to the desired values, is described. Finally, it is explained why this process is appropriate in a BELOCA analysis performed with the ASTRUM methodology.

1.0 ASTRUM Analysis Axial Power Shape Generation Process

As shown in Reference 2 Table 1-10, as part of an ASTRUM uncertainty analysis, the following parameters are among the independently sampled uncertainty contributors:

- [

  ]a,c
- [

  ]a,c
- [ ]a,c
- [

  ]a,c

For each of the 124 cases in an ASTRUM uncertainty analysis, [

]a,c

[

]^{a,c} The axial power shape input affects the maximum initial stored energy in the hot rod, and the axial distribution of initial stored energy, which are important initial conditions for the LOCA transient. Also, in the ASTRUM EM, [

]^{a,c}

Subsequently, each WCOBRA/TRAC steady-state output is inspected to assure that [ ]^{a,c} in compliance with the ASTRUM EM (see Reference 2 Table 12-6). As discussed, cases with PLHR higher than desired may be accepted as more conservative than required by the EM. [

]^{a,c}

2.0 Process for Axial Power Shape Adjustments in Limiting CRR Runs

2.1 Selection of Cases for Axial Power Shape Modification

When the CRR ASTRUM analysis was executed, it was noted that [ ][a,c] ASTRUM cases had calculated FQ values (and therefore peak linear heat rates) higher than the desired values based on the sampled parameters for each run. [

][a,c] These cases included the limiting case in the ASTRUM analysis, run031, which had an axial power shape with FQ 15% higher than the sampled value for that case, and a correspondingly high PLHR. Similarly, the 2nd highest PCT case, run087, had an axial power shape with FQ 12% higher than the sampled value. [

][a,c]

When the impact of TCD on the **AP1000** PWR BELOCA analysis PCT result was to be quantified, the higher FQ values in the CRR analysis limiting cases were identified as an analysis margin which could partially offset the PCT effect of TCD. The approach to quantify this analysis margin was [

][a,c]

2.2 Process for Axial Power Shape Adjustment

[

][a,c]

[ ]^{a,c} Therefore, for quantification of the CRR analysis margin, PBOT and PMID input were modified to generate revised axial power shapes for the top two PCT cases.

[

]^{a,c}

The PBOT, PMID modifications for the top two CRR analysis cases are summarized in Table 1. The original CRR and revised axial power shapes are shown in Figure 1 and Figure 2 for run031 and run087, respectively. The as-sampled FQ values and the as-executed FQ values for these runs in the CRR and the margin cases are summarized in Table 2. Similarly, the desired PLHR and as-executed PLHR values for these runs are summarized in Table 3. The PBOT/PMID space with the original sampled points for all 124 runs and the modified points for the top 2 cases are shown in Figure 3. The revised axial power shapes meet the criteria specified above.

2.3 Calculation Results with Adjusted Axial Power Shape

When the CRR analysis run031 and run087 were executed with the revised axial power shape input, [

]^{a,c}

Selected plots for run031 with the original CRR analysis axial power shape and with the modified axial power shape are shown in Figure 4 through Figure 7. [

]^{a,c}

[

]^{a,c}

3.0 Effect of PBOT/PMID Adjustments on ASTRUM Analysis Results

In the ASTRUM EM, [

]^{a,c}

[

]^{a,c}

Relationship to Other CRR RAI Responses

[

]^{a,c}

RAI CRR-008 requests execution of the full set of 124 ASTRUM analysis cases explicitly considering TCD. The same axial power shape modifications described in Section 2 for run031 and run087 will be applied in execution of the full set of 124 runs. Similar axial power shape modifications will be made for other cases as necessary [

]^{a,c} For each case where modification of the axial power shape is performed, the process described in Section 2.2 will be followed.

Table 1: PBOT/PMID Input Updates for CRR Analysis PLHR Margin Cases

a,c

Table 2: Sampled and As-Executed FQ Values for CRR Analysis and PLHR Margin Cases

a,c

Table 3: Desired and As-Executed PLHR Values for CRR Analysis and PLHR Margin Cases

a,c

Table 4: Results of CRR Analysis PLHR Analysis Margin Cases

a,c

a,c

Figure 1: Hot Rod Axial Power Shape Comparison for CRR Run031

a,c

Figure 2: Hot Rod Axial Power Shape Comparison for CRR Run087

a,c

Figure 3: CRR Original Sampled PBOT, PMID Points and Location of Modified Points for Top 2 PCT Cases

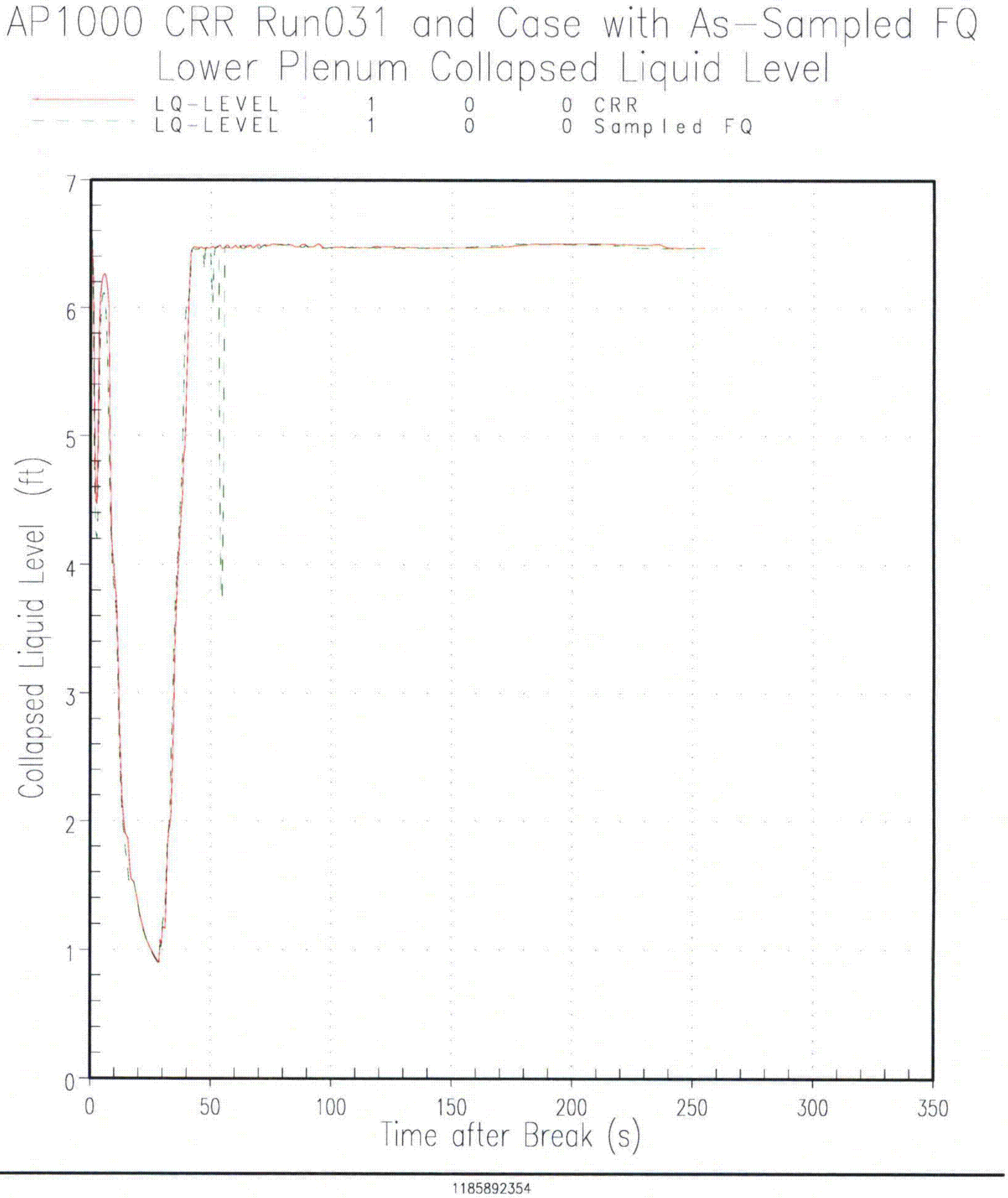

Figure 4: **AP1000** CRR Run031 Impact of Axial Power Shape Change: Lower Plenum Collapsed Liquid Level

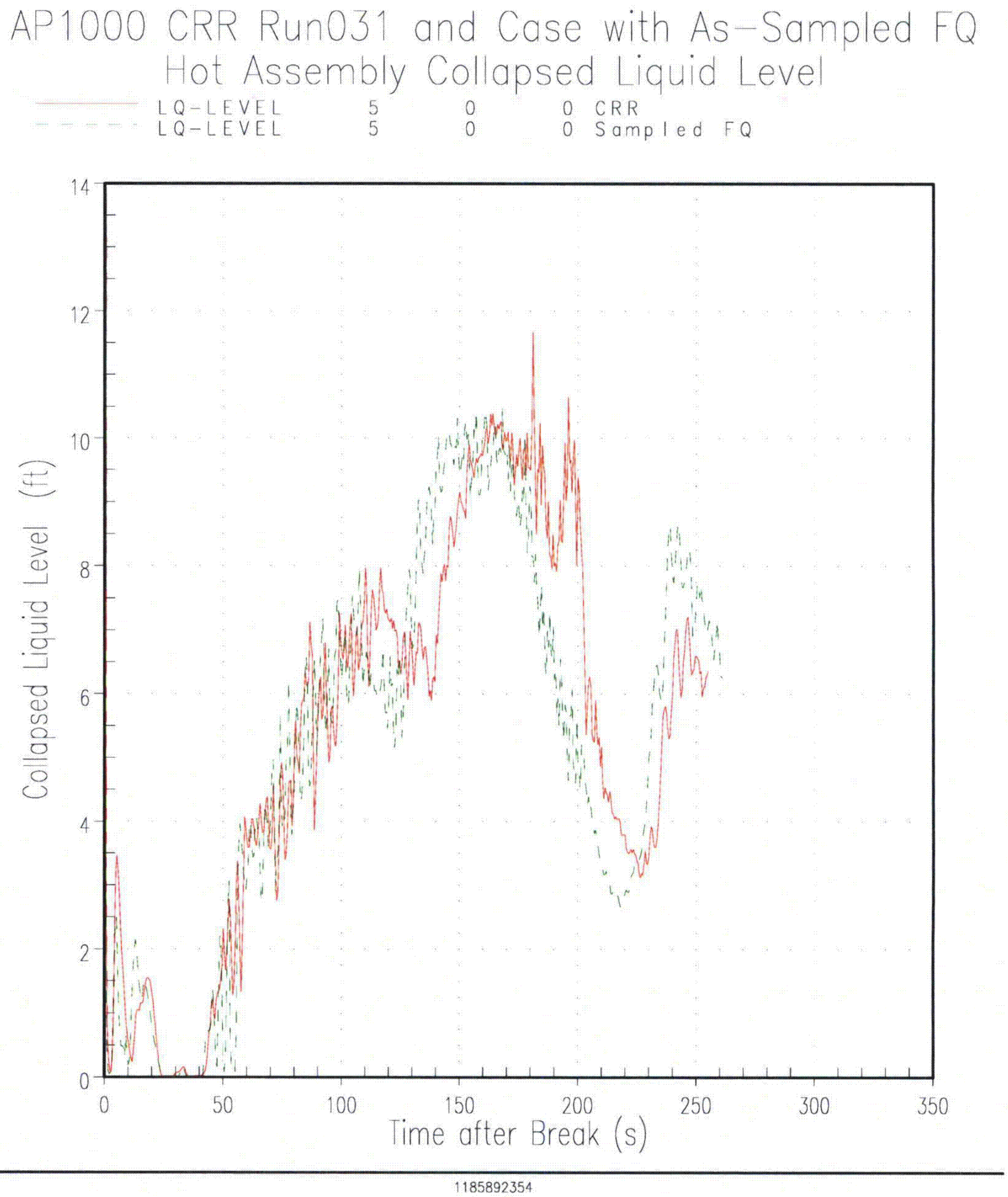

Figure 5: **AP1000** CRR Run031 Impact of Axial Power Shape Change: Hot Assembly Channel Collapsed Liquid Level (from bottom of active fuel)

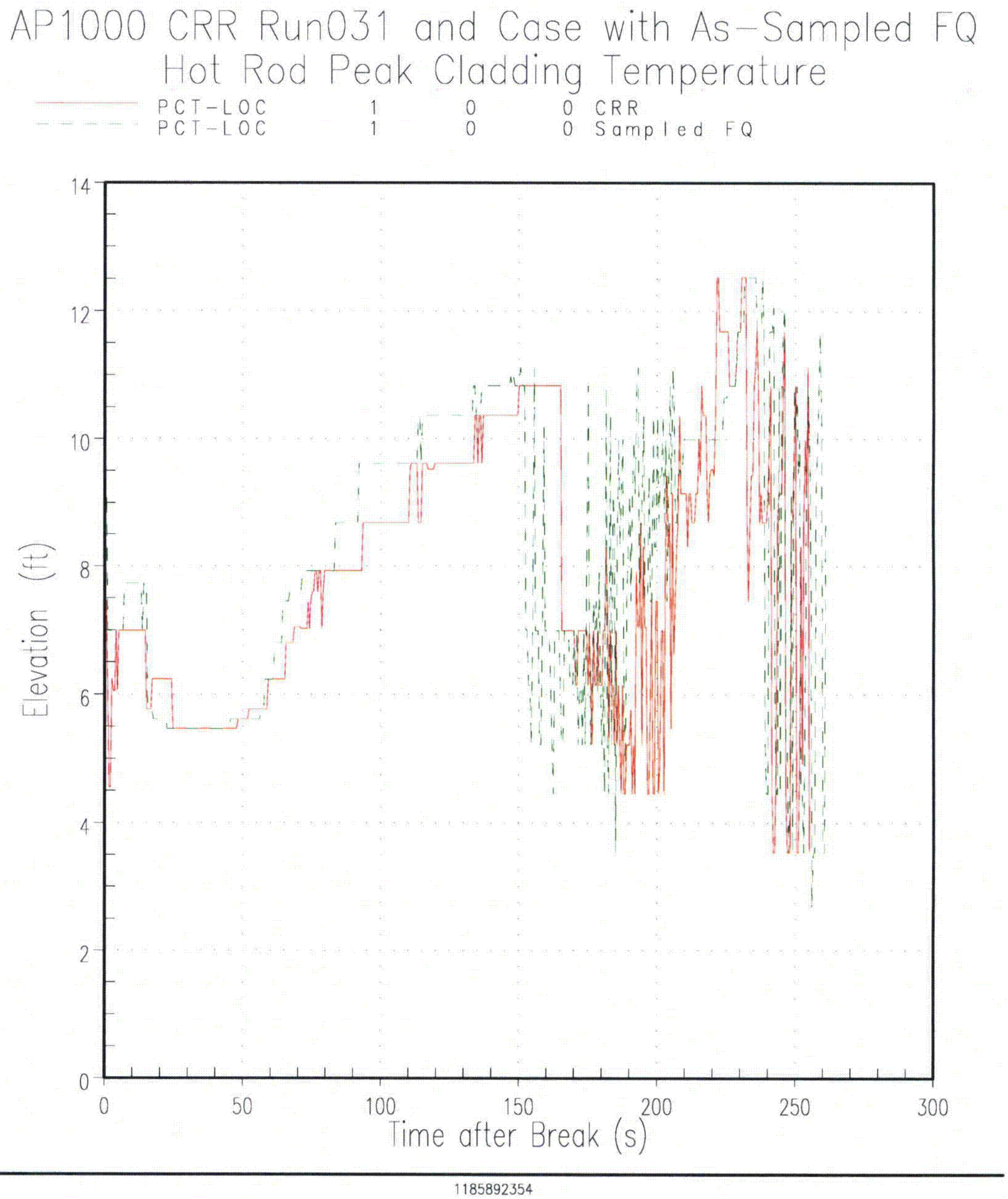

Figure 6: **AP1000** CRR Run031 Impact of Axial Power Shape Change: WCOBRA/TRAC Hot Rod Peak Cladding Temperature Location

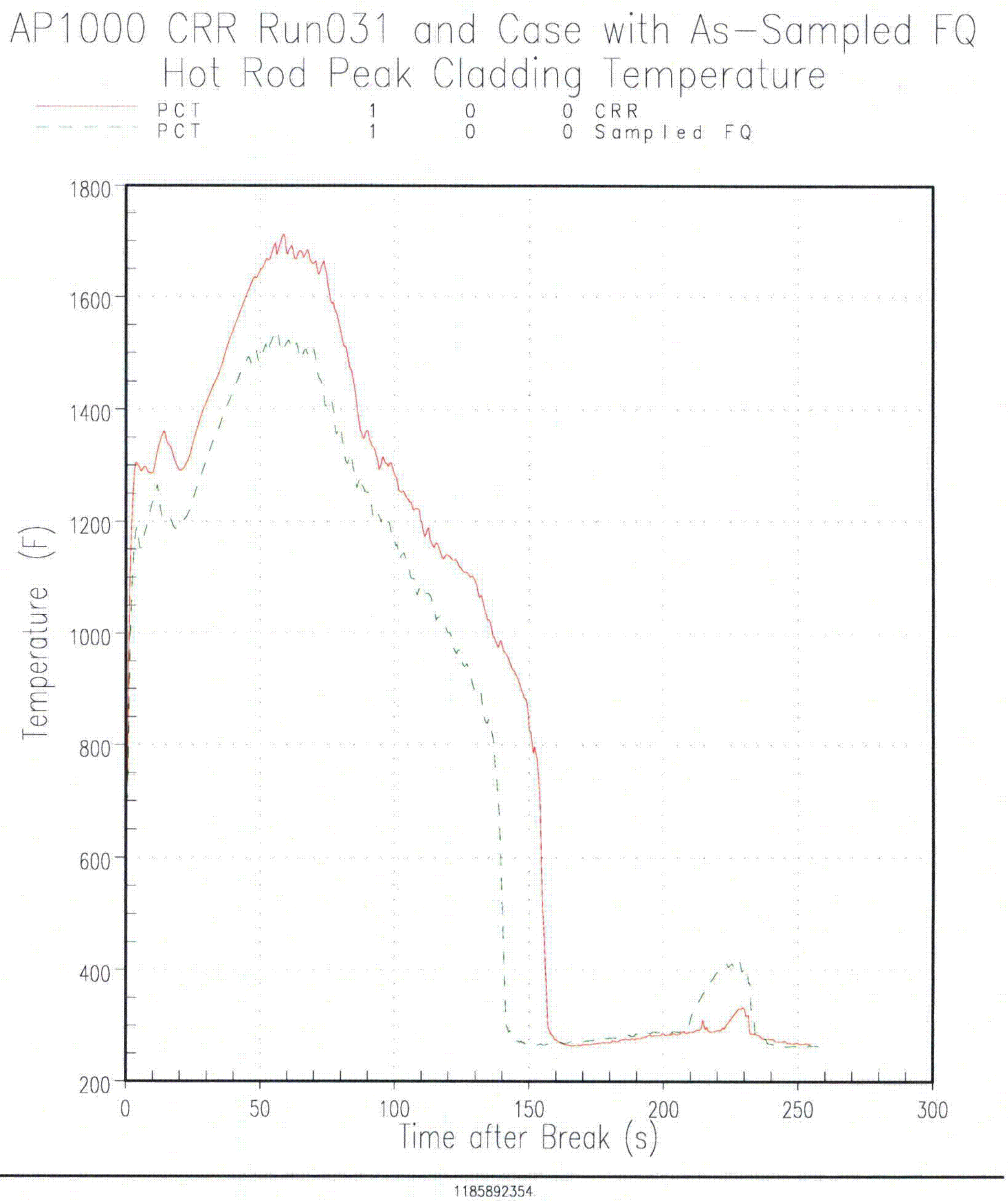

Figure 7: **AP1000** CRR Run031 Impact of Axial Power Shape Change: WCOBRA/TRAC Hot Rod Peak Cladding Temperature

References

1. WCAP-17524-P, "AP1000 Core Reference Report," March 2012

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

3. APP-SSAR-GSC-772, "Evaluation of Thermal Conductivity Degradation for AP1000 AFCAP and DCD Rev. 19 Best Estimate Large Break LOCA ASTRUM Analyses," June 2012

4. LTR-NRC-12-46, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to Address Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," June 13, 2012

Appendix 1. Description of Power Shape Generation Using the PSHAPE Code

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

a,c

[ ]^{a,c}

[

]a,c

- [

]^{a,c}

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]^{a,c}

- [

]^{a,c}

a,c

[

]^{a,c}

CRR-006 *(LBLOCA Analysis)*

Supplemental Information to WCAP-17524 (LTR-NRC-12-46 P-Attachment, June 2012) states that the large-break LOCA analysis evaluation of the fuel TCD effects considered peaking factor burndown effects. Table 5-2, "Peaking Factor Assumed in the Evaluation of TCD," of the Supplemental Information provides the values of the peaking factors (F_Q and FDH) as a function of rod burnup.

Describe how the values of F_Q and F_{DH} in Table 5-2 are determined and how they are implemented in the large-break LOCA analysis.

Westinghouse Response to CRR-006

Values of F_Q and F_{DH} were calculated utilizing the Westinghouse Advanced Nodal Code (ANC). ANC is a highly accurate and efficient three dimensional (3-D) spatial code which is licensed to confirm peaking factors for the **AP1000** Pressurized Water Reactor (PWR) safety analysis. For the thermal conductivity degradation (TCD) evaluation, representative **AP1000** PWR cycle designs were modeled using ANC.
[

]^{a,c}

Once maximum peaking factors were calculated, bounding limits could be determined utilizing the F_Q and F_{DH} data points as a function of rod burnup. These bounding limits are equivalent to the values used in the TCD evaluation. Note that the Core Reference Report (CRR, WCAP-17524-P) F_Q and F_{DH} peaking factor limits were utilized for the first 30 GWd/MTU of rod burnup. At rod burnups beyond 30 GWd/MTU, bounding limits could be reduced due to peaking factor burndown. [

]^{a,c}

Physically accounting for TCD leads to an increase in fuel temperature as the fuel is burned. At the same time, accounting for peaking factor burndown leads to a relative reduction in fuel temperature. [

]^{a,c}

When TCD and peaking factor burndown are explicitly considered, the assembly with the highest operating power may or may not also have the highest stored energy. In the operating core, the burnup and power history of the assembly/rod with the highest stored energy at any given time is a function of the specific core design and operating history. [

]^{a,c}

The reduced peaking factors at high burnup conditions used in the Large Break LOCA analysis constitute a core design constraint and will be confirmed to be met during the future reload process, similar to other limits.

References

1. APP-GW-GL-700 Revision 19, "Design Certification Document," June 2011.

2. WCAP-17524-P, "AP1000 Core Reference Report," March 2012.

3. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

CRR-007 *(LBLOCA Analysis)*

Calculation Note APP-SSAR-GSC-772 indicated that the large-break LOCA analysis for the evaluation of the fuel TCD effect is performed using RUV reactor coolant pump designed by KSB.

(a) Provide a list of significant differences or changes in the LBLOCA-TCD WCOBRA/TRAC analysis from the analysis in the AP1000 Design Control Document.

(b) Describe why each of these changes is necessary and appropriate.

Westinghouse Response to CRR-007

The **AP1000** PWR best estimate large break loss of coolant accident (BELOCA) analysis presented in the Core Reference Report (Reference 1), and the prior BELOCA analysis presented in the Design Control Document (DCD) Revision 19 (Reference 3) were performed using the ASTRUM evaluation model (EM) (Reference 2). Application of the ASTRUM EM for **AP1000** PWR BELOCA analysis was approved as part of the certification of DCD Revision 19 (see Reference 4).

Neither the BELOCA analysis presented in the DCD Revision 19, nor the analysis presented in the CRR were performed using the RUV reactor coolant pump designed by KSB. These analyses are not applicable to **AP1000** plants built with KSB pumps without additional evaluation. Therefore, the BELOCA evaluation of fuel TCD effects is not applicable to **AP1000** plants built with KSB pumps.

The primary differences between the **AP1000** plant WCOBRA/TRAC model and ASTRUM uncertainty analysis in the DCD Revision 19, and the updated WCOBRA/TRAC model and ASTRUM uncertainty analysis in the as-submitted CRR are:

(1) Changes to the fuel assembly mechanical design features (primarily the grid design)
(2) Changes to the reactor coolant pump design and homologous curves
(3) Changes to the upper head structures
(4) Increase in the time before reactor coolant pump trip following a LBLOCA when offsite power is available
(5) Changed PBOT/PMID box
(6) Changed FdH limit

Items (1), (5) and (6) were necessary to support the advanced first core fuel assembly and core design as described in the CRR.
Item (4) was necessary for consistency with the updated non-LOCA analyses presented in the CRR.

Item (2) was necessary to incorporate modified reactor coolant pump parameters resulting from design finalization.
Item (3) was necessary to incorporate the upper head structure design which is shown in the DCD Figure 3.9-6, as part of providing an integrated response to design changes resulting from design finalization of the **AP1000** PWR.

Therefore, incorporation of these changes into the **AP1000** plant WCOBRA/TRAC model and ASTRUM uncertainty analysis was necessary and appropriate to support the modified plant, fuel assembly, and core designs.

The HOTSPOT code version used in the BELOCA analysis presented in the CRR was version 8.0, while version 6.1 was used in the BELOCA analysis presented in the DCD Revision 19. The change in HOTSPOT code version reflects the latest released code versions at the time the analyses were executed. Westinghouse has policies and procedures in place for evaluation and reporting of the effects of computer code changes on LOCA PCT results in compliance with 10 CFR 50.46. The HOTSPOT code changes between version 6.1 and version 8.0 have been considered for reporting under 10 CFR 50.46. For the **AP1000** plant analyses, the reportable changes fall into the following categories:

- General code maintenance
- Non-discretionary changes with an estimated PCT impact of 0°F
- Non-discretionary change which has been adequately incorporated into the TCD evaluation submitted for the certified design (Reference 6), leading to an estimated PCT impact of 0°F.

Fuel pellet thermal conductivity degradation (TCD) and peaking factor burndown were not explicitly considered in the as-approved ASTRUM EM. To demonstrate that the **AP1000** plant continues to meet the acceptance criteria specified by 10 CFR 50.46 (b)(1), (b)(2), and (b)(3), changes were made to the ASTRUM EM to explicitly consider and adequately account for the burnup-dependent aspects of the fuel performance changes considering TCD. Specifically, the following changes were made:

- Fuel performance data which explicitly accounts for the effects of TCD was used as input, as described in LTR-NRC-12-46 (Reference 5). This is different from the as-approved EM which was licensed using PAD 4.0.
- [

  ]^{a,c}
- As described in Reference 5, a thermal conductivity model appropriate for the TCD evaluations was incorporated into WCOBRA/TRAC and HOTSPOT. This is different from the thermal conductivity model in the as-approved EM.
- [

  ]^{a,c}

The changes described above are applicable for the **AP1000** plant BELOCA analysis which explicitly considers TCD as described in response to RAI CRR-008.

References

1. WCAP-17524-P, "AP1000 Core Reference Report," March 2012.

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

3. APP-GW-GL-700 Revision 19, "Design Certification Document," June 2011.

4. NUREG-1793 Supplement 2, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design," August 2011. NRC ADAMS accession number ML112061231.

5. LTR-NRC-12-46, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report,' to address Thermal Conductivity Degradation (Proprietary/Non-Proprietary)'," June 13, 2012.

6. DCP_NRC_003214, "10 CFR 50.46 Thirty (30) Day Report for the AP1000® Standard Plant Design," June 13, 2012.

CRR-008 *(LBLOCA Analysis)*

To properly address fuel thermal conductivity degradation effects on the large-break LOCA analysis for the advanced first core application program, Westinghouse should perform reanalysis of the best-estimate large-break LOCA analysis with total number of runs required by the ASTRUM evaluation method.

Westinghouse Response to CRR-008

The **AP1000** plant best-estimate large-break loss-of-coolant accident (BELOCA) analysis presented in the Core Reference Report (Reference 1) was performed using the ASTRUM evaluation model (EM) (Reference 2). Application of the ASTRUM EM for **AP1000** plant BELOCA analysis was approved as part of the certification of Design Control Document (DCD) Revision 19 (Reference 3, see the Reference 7 submittal and Reference 4 Safety Evaluation Report (SER)).

The ASTRUM EM is based on the PAD 4.0 fuel performance code (Reference 5). PAD 4.0 was licensed without explicitly considering fuel thermal conductivity degradation (TCD) with burnup, or peaking factor burndown. Explicit modeling of TCD in the fuel performance code leads directly to increased fuel temperature (pellet radial average temperature) as well as other fuel performance-related effects beyond beginning-of-life. Since PAD provides input to the BELOCA 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) and oxidation if there is no provision to credit off-setting effects.

To demonstrate that the **AP1000** plant continues to meet the acceptance criteria specified by 10 CFR 50.46, a plant-specific adaptation of the ASTRUM EM was applied to explicitly consider the burnup-dependent aspects of the fuel performance changes considering TCD.

The plant-specific adaptation of the ASTRUM uncertainty methodology is described and results of the analysis are summarized. Updates to the description of the **AP1000** plant LBLOCA response in the CRR Appendix F are provided in Appendix 1 of this response.

1.0 Plant-Specific Adaptation of ASTRUM EM for *AP1000* Plant BELOCA Analysis Considering TCD

The ASTRUM EM analysis process is presented for a sample plant in Section 12 of Reference 2 and is based on the following steps:

1. Plant description and nodalization (Reference 2 Section 12-2)
2. Development of the reference transient and allowable plant operating ranges (Reference 2 Section 12-3)
3. Execution and analysis of the reference transient (Reference 2 Section 12-4)
4. Development of the ASTRUM run matrix (Reference 2 Section 12-5)
5. ASTRUM analysis execution, results, determination of the 95/95 singular uncertainty statement and compliance with 10 CFR 50.46 criteria (Reference 2 Section 12-6 and 12-7)

The plant-specific adaptation of the ASTRUM EM for **AP1000** plant analysis considering TCD is described, considering each of these steps.

1.1 Plant Description and Nodalization

The **AP1000** plant description and WCOBRA/TRAC model nodalization are unchanged from the ASTRUM analysis approved in DCD Revision 19 (Reference 3).

Response to CRR-007 describes plant design changes which were incorporated in the CRR ASTRUM analysis considering TCD.

1.2 Development of the Reference Transient and Allowable Plant Operating Ranges

The development of the reference transient and allowable plant operating ranges is unchanged from the as-approved ASTRUM EM.

As stated in Reference 2 Section 12-3-6, several of the reference conditions bounded in the uncertainty analysis are determined on a plant-specific basis. The conservative values for these parameters are determined [

]^{a,c} The list of these parameters is presented in Section 11-3-1 of Reference 2, which includes:

- Steam generator tube plugging level (SGTP)
- Offsite power availability (LOOP/OPA)
- Peripheral assembly average power (PLOW)

[

]^{a,c}

The **AP1000** plant ranges for these parameters are not affected by TCD. [

]^{a,c}

1.3 Execution and Analysis of the Reference Transient

The execution and analysis of the Reference Transient are unchanged from the as-approved ASTRUM EM.

[

]^{a,c}

In compliance with the **AP1000** PWR SER requirements, sensitivity calculations are performed to quantify the PCT effect of assuming the core makeup tanks (CMTs) or passive residual heat removal heat exchanger (PRHR) is inoperable. [

]^{a,c} The 2°F penalty is applied as a bias to the 95/95 PCT estimate results of the uncertainty analysis considering TCD.

1.4 Development of the ASTRUM Run Matrix

The development of the ASTRUM run matrix is unchanged from the as-approved ASTRUM EM except for:

- Changes made to account for the burnup-dependent nature of the integral effects of TCD and peaking factor burndown, as described in response to CRR-006
- Modification of the axial power shape PBOT and/or PMID input in some cases as described in response to CRR-005

In the as-approved ASTRUM EM, the average and low power assemblies [

]^{a,c} This approach for the treatment of the average and low power assembly burnups is justified as discussed in response to CRR-004.

The process utilized to generate the ASTRUM run matrix is described in Reference 2 Section 11-1-2. For the **AP1000** plant ASTRUM uncertainty analysis modeling TCD and peaking factor burndown, [

]^{a,c}

[

]^{a,c}

1.5 ASTRUM Analysis Execution, Results, Determination of the 95/95 Singular Uncertainty Statement and Compliance with 10 CFR 50.46

For the plant-specific adaptation of the ASTRUM EM applied to the **AP1000** plant, changes were made in execution of the uncertainty analysis calculations to explicitly consider the burnup-dependent aspects of the fuel performance changes considering TCD. First, the ASTRUM run matrix was developed as previously described. In addition, the following changes were made in execution of the Cycle 1 and Cycle 2 calculations (as described in response to CRR-007):

- Fuel performance data which explicitly accounts for the effects of TCD were used as input. The fuel performance data were generated for the **AP1000** plant as described in LTR-NRC-12-46 (Reference 6).
- As described in Reference 6, a thermal conductivity model which accounts for TCD was incorporated into WCOBRA/TRAC and HOTSPOT. This thermal conductivity model was used in the uncertainty analysis calculations.
- [

  ]^{a,c}

Consistent with the ASTRUM EM, the limiting PCT, MLO and CWO results from the calculations are the 95/95 estimators for the plant analysis and are compared to the acceptance criteria specified by 10 CFR 50.46.

10 CFR 50.46 requires a "high level of probability" that the acceptance criteria would not be exceeded. The ASTRUM SER stated that conformance to this requirement was achieved to satisfaction with a 95/95 tolerance limit jointly on the three criteria (PCT, MLO and CWO). However, it is also recognized that the ASTRUM EM retains an additional layer of conservatism in both the EM and the inputs, which enhance the confidence further, significantly beyond the 95% level. Also, a joint probability tolerance based on three outcomes is quite conservative as it ignores correlation between PCT and MLO. Moreover, the 3rd criterion (CWO) is typically shown to satisfy the acceptance criteria with ample margin.

2.0 *AP1000* Plant BELOCA Analysis Results Considering TCD

The **AP1000** plant CRR ASTRUM uncertainty analysis was revised to explicitly consider the burnup-related effects of TCD. The revised uncertainty analysis was executed following the plant-specific adaptation of the ASTRUM EM described in Section 1.0.

For the revised uncertainty analysis, the peaking factor burndown assumed is summarized in Table 1. The burndown after 55 GWd/MTU is different than presented in Reference 6; the additional burndown towards end of life better reflects the power and peaking limits expected for higher burnup assemblies. As noted in response to CRR-006, the peaking factor burndown limits assumed in the LBLOCA analysis

constitute a core design constraint and will be confirmed to be met during the future reload process, similar to other limits.

The limiting PCT, MLO and CWO results of the uncertainty analysis are summarized in Table 2.

3.0 10 CFR 50.46 Requirements

It must be demonstrated that there is a high level of probability that the following limits set forth in 10 CFR 50.46 are not exceeded:

(b)(1) The limiting PCT from the revised BELOCA analysis considering TCD corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Because the resulting PCT for the limiting case is 1936°F for the **AP1000** plant (including 2°F bias for sensitivity to PRHR inoperable), the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), that is, "Peak Clad Temperature less than 2200°F," is demonstrated. This result is shown in Table 2.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Because the resulting MLO for the limiting case is 4.2% for the **AP1000** plant, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), that is, "Maximum Local Oxidation of the cladding less than 17 percent," is demonstrated. This 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. While the limiting MLO is determined based on the single Hot Rod, the CWO value can be conservatively chosen as that calculated for the limiting Hot Assembly Rod (HAR) when there is significant margin to the regulatory limit. The limiting HAR total maximum oxidation is 0.3% for the **AP1000** plant. Thus, a detailed CWO calculation is not needed because the calculations would include many lower power assemblies and the outcome would be less than the limiting HAR total maximum oxidation. Therefore, this analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), that is, "Core-Wide Oxidation less than 1 percent," is demonstrated. This 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 are met. The approved methodology specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low power peripheral channel as defined in the WCOBRA/TRAC model. This situation has not been calculated to occur for the **AP1000** plant. Therefore, acceptance criterion (b)(4) is 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 emergency core cooling system (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 in the **AP1000** plant to maintained long-term cooling remain unchanged with the application of the plant-specific modification of the ASTRUM EM considering TCD.

The **AP1000** plant passive core cooling system provides effective core cooling following a large-break LOCA event. The large-break LOCA transient for the limiting PCT case has been extended beyond fuel rod quench to the time at which the CMT liquid level has decreased to the setpoint that activates the fourth-stage ADS valves and IRWST injection. A significant increase in safety injection flow rate occurs when the IRWST becomes active. The analysis performed demonstrates that CMT injection is sufficient to maintain the mass inventory in the core and downcomer, from the period of fuel rod quench until IRWST injection. The **AP1000** plant passive core cooling system provides effective post-LOCA long-term core cooling; therefore, acceptance criterion (b)(5) is satisfied.

Based on the BELOCA analysis results using a plant-specific adaptation of the ASTRUM EM to account for burnup-dependent effects of TCD (see Table 2), it is concluded that the **AP1000** plant continues to maintain a margin of safety within the limits prescribed by 10 CFR 50.46.

Table 1. Summary of Peaking Factor Burndown Supported by **AP1000** Plant Best Estimate Large Break LOCA CRR Updated Analysis Considering TCD

Hot Rod Burnup (GWd/MTU)	FdH (includes uncertainties)[1]	FQ Transient (Max FQ, includes uncertainties)	FQ SS Baseload (without uncertainties)
0	1.72	2.60	2.10
30	1.72	2.60	2.10
49	1.55	2.30	1.85
55	1.55	2.30	1.85
62	1.40	1.90	1.45
1. Hot assembly power follows the same burndown, since it is a function of FdH			

Table 2: AP1000 Plant Best Estimate Large Break LOCA CRR Updated Analysis Considering TCD - Comparison of Results to Current 10 CFR 50.46(b) Acceptance Criteria

	Result	Acceptance Criteria
95/95 PCT [1]	1936°F [4]	< 2200°F
95/95 Transient MLO [2]	4.2%	< 17%
95/95 CWO [3]	0.30%	< 1%
Coolable Geometry	Criterion met	Remains coolable
Long-Term Cooling	Criterion met	Long term cooling provided
1. Peak Cladding Temperature 2. Maximum Local Oxidation 3. Core-wide Oxidation 4. Includes 2°F penalty for sensitivity to PRHR inoperable		

References

1. WCAP-17524-P, Revision 0, "AP1000 Core Reference Report," March 2012.

2. WCAP-16009-P-A, Revision 0, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

3. APP-GW-GL-700, Revision 19, "AP1000 Design Certification Document," June 2011.

4. NUREG-1793 Supplement 2, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design," August 2011. NRC ADAMS accession number ML112061231.

5. WCAP-15063-P-A, Revision 1, with Errata, "Westinghouse Improved Performance analysis and Design Model (PAD 4.0)," 2000.

6. LTR-NRC-12-46, Revision 0 "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report,' to address Thermal Conductivity Degradation (Proprietary/Non-Proprietary)'," June 13, 2012.

7. APP-GW-GLE-026, Revision 1, "Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000," January 2009.

Appendix 1. Markups for CRR Appendix F due to Revised BELOCA Analysis

Markups to the applicable pdf pages of the CRR are provided, followed by the identified text inserts, updated table, and updated figures.

Table 15.0-2 (Sheet 5 of 5)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density (Δk/gm/cm^3)	Moderator Temperature (pcm/°F)	Doppler	
15.6	Decrease in reactor coolant inventory					
	Inadvertent opening of a pressurizer safety valve and inadvertent operation of ADS	LOFTRAN	0.0	–	Upper curve of Figure 15.0.4-1	3415
	Steam generator tube failure	LOFTTR2	0.0	–	Lower curve of Figure 15.0.4-1	3449.15 (a)
	A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate containment	NA	NA	–	NA	NA
	LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	NOTRUMP <u>W</u>COBRA/ TRAC	See subsection 15.6.5 references	–	See subsection 15.6.5 references	3434.0 (a) (b)

Notes:

a. The Non LOCA analyses assume an initial power of 101% of the NSSS Power (NSSS Power = rated thermal power (RTP) plus 15 MWt for pump heat) and the LOCA analyses assume an initial power of 101% of RTP.

(b) Section 15.6.5.4A describes the large-break LOCA analysis methodology, which includes treatment of the initial thermal power output uncertainty.

streams out the top of the containment shield building and is reflected back down by air-scattering. The total of the three dose paths is within the dose criteria of 5 rem TEDE as defined in GDC 19.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and is reported in Table 15.6.5-3.

15.6.5.4 Core and System Performance

Subsection 15.6.5.4A describes the large-break LOCA analysis methodology and results. Subsections 15.6.5.4B.1.0 through 15.6.5.4B.4.0 describe the small-break LOCA analysis methodology and results.

15.6.5.4A Large-break LOCA Analysis Methodology and Results

Westinghouse applies the WCOBRA/TRAC computer code to perform best-estimate large-break LOCA analyses in compliance with 10 CFR 50 (Reference 5). WCOBRA/TRAC is a thermal-hydraulic computer code that calculates realistic fluid conditions in a PWR during the blowdown and reflood of a postulated large-break LOCA. The methodology used for the AP1000 analysis is documented in WCAP-12945-P-A, WCAP-14171, Revision 2, ~~and~~ WCAP-16009-P-A (References 10, 11, ~~and~~ 32). , and Reference 31

ASTRUM

The NRC staff has reviewed and approved ~~the~~ best-estimate LOCA methodology (ASTRUM methodology), as documented in the SER attached in front of Reference 32 for estimating the 95th percentile PCT for two-loop, three-loop and four-loop Westinghouse pressurized water reactors (PWRs) and the AP600. ~~In Reference 3, the NRC staff has reviewed and approved a best-estimate LOCA methodology, as documented in Reference 11, for estimating the 95th percentile PCT for the AP600. In the Reference 32 and Reference 11 methodologies,~~ the WCOBRA/TRAC code is used to calculate the effects of initial conditions, power distributions, and global models, and the HOTSPOT code is used to calculate the effects of local models.

Replace text with Insert A

References 34 and 31

In the ASTRUM ~~uncertainty~~ methodology (Reference 32), as used in the AP1000 LB LOCA analysis, global models and ~~initial condition~~, power-distribution, and local uncertainties are sampled independently for each of 124 runs over ~~the~~ same ranges of uncertainty and distributions as in References 10, 32, and 33, as described in ~~Reference 34~~. The sampled global models, initial conditions, and power-distribution uncertainties become inputs to each of the ~~124~~ WCOBRA/TRAC calculations. The thermal-hydraulic boundary conditions for the hot rod are input to the local uncertainties calculation performed by the HOTSPOT code.

Results from the ~~124~~ calculations are ranked by PCT from highest to lowest. A similar procedure is repeated for maximum local oxidation (MLO) and core wide oxidation (CWO). In order statistics as applied in the ASTRUM methodology, the limiting case for a parameter, such as peak cladding temperature (PCT), is a conservative estimate of the 95th percentile with 95 percent confidence. The limiting PCT, limiting MLO, and CWO may come from the same case or as many as three different cases because each parameter is assumed to be independent of the other two. The assumption of independence of the calculated licensing parameters is a conservative assumption because there is a dependence of MLO and CWO on cladding temperature.

Replace text with Insert B

~~For the AP1000 large-break LOCA analysis, the best-estimate LOCA analysis methodology is applied as described in Reference 34. The best-estimate large-break LOCA analysis complies with the stipulated applicability limits in the Reference 3 approval and the Reference 32 approval.~~ The post-LOCA long-term core cooling and core boron concentration analyses discussed in subsection 15.6.5.4C are applicable to the large-break LOCA transient.

15.6.5.4A.1 General Description of WCOBRA/TRAC Modeling

WCOBRA/TRAC is the best-estimate thermal-hydraulic computer code used to calculate realistic fluid conditions in the PWR during blowdown and reflood of a postulated large-break LOCA.

The WCOBRA/TRAC Code Qualification Document (Reference 10) contains a complete description of the code models and justifies their applicability to PWR large-break LOCA analysis.

Table 15.6.5-4 lists AP1000-specific parameters identified for use in the large-break LOCA analysis. WCOBRA/TRAC studies were performed for AP1000 to establish sensitivities to parameter variations. These studies included effects of ranging steam generator tube plugging, ranging the relative power in the low-power assemblies, loss of offsite power coincident with the break initiation, and break location. The calculated results were used to identify bounding conditions, which are then used in the AP1000 uncertainty calculations.

The WCOBRA/TRAC vessel nodalization is developed from plant design drawings to divide the vessel into 10 vertical sections. The bottom of section 1 is the inside vessel bottom, and the top of section 10 is the inside top of the vessel upper head. In addition to the major downcomer and core flow paths, the modeled bypass flow paths are the upper head cooling spray, guide thimbles, and core bypass. After defining the elevations for each section, a noding scheme is defined for the WCOBRA/TRAC model as shown in Reference 34. WCOBRA/TRAC assumes a vertical flow path for vertically stacked channels, unless specified otherwise in the input. Positive flow

for the vertically connected channels (and cells) is upward. Several of the 10 sections are divided vertically into 2 or more levels; these levels are referred to as cells within a channel.

The WCOBRA/TRAC loop model represents the major primary, secondary, and passive safety systems components. Both loops are explicitly modeled, including the hot leg, the steam generator, and the two cold legs and associated pumps. The loop designated "1" has the pressurizer and the PRHR system connections, and loop "2" cold legs have the core makeup tank pressure balance line connections. The reactor coolant pump models contain the AP1000 homologous curves together with appropriate two-phase head and torque multipliers and degradation data. AP1000 values for pump coastdown characteristics are also applied. The passive safety features are modeled using design data for elevations, liquid volumes, and line losses. Because the ADS is not actuated until long after the time of PCT in large-break LOCA events, it is not modeled in detail.

15.6.5.4A.2 Steady-state Calculation

A WCOBRA/TRAC LOCA calculation is initiated from a point at which the flows, temperatures, powers, and pressures are at their approximate steady-state values before the postulated break occurs. Steady-state WCOBRA/TRAC calculations are run for a brief time period to verify that the calculated conditions are steady and that the desired reactor conditions are achieved.

, accounting for the effects of thermal conductivity degradation as described in Reference 31

The values used to set the steady-state plant conditions reflect the AP1000 parameters for reactor coolant pump flows, core power, and steam generator tube plugging levels. The fuel parameters provide the steady-state fuel temperatures, pressures, and gap conductances as a function of fuel burnup and linear power. The calculated fuel temperatures from WCOBRA/TRAC are adjusted to match the specified fuel data by adjusting the gap heat transfer coefficient between the pellet and the cladding. Once the vessel fluid temperatures, flows, pressures, loop pressure drop, and core parameters are in agreement with the desired values and are steady, a suitable initial condition is achieved.

15.6.5.4A.3 Signal Logic for Large-break LOCA

The reactor trip signal occurs due to compensated pressurizer pressure within the first seconds of the large-break transient however control rod insertion is not modeled in WCOBRA/TRAC and no effects of control rod insertion on reactivity ensue. A safeguards "S" signal occurs due to containment high pressure of 6.7 psig at 2.2 seconds of large-break LOCA transients.

As a consequence of this signal, after appropriate delays, the PRHR and core makeup tank isolation valves open, containment isolation occurs, and the reactor coolant pump automatic trip timer begins. The rapid depressurization of the primary system during a large-break LOCA leads to the initiation of accumulator injection early in the large-break transient. The accumulator flow

diminishes core makeup tank delivery to such an extent that the core makeup tank level does not approach the ADS Stage 1 valve actuation point until after the accumulator tank is empty. The accumulator empties long after the blowdown portion of the large-break LOCA transient is complete. Actuation of the ADS on CMT water level does not occur until long after the AP1000 PCT is calculated to occur.

15.6.5.4A.4 Transient Calculation

Once the steady-state calculation is found to be acceptable, the transient calculation is initiated. The semi-implicit pipe break model is added to the desired break location. Cold-leg breaks are analyzed because the hot-leg break location is nonlimiting in the large-break LOCA best-estimate methodology. The break size and type are sampled consistent with the WCAP-16009-P-A (Reference 32) methodology. The containment backpressure is specified consistent with WCAP-16009-P-A (Reference 32) methodology. The steady-state calculation is restarted with the above changes to begin the transient.

Table 15.6.5-5 shows a general sequence of events following a large cold-leg break LOCA and the relationship of these events to the blowdown and reflood portion of the transient.

15.6.5.4A.5 Large-break LOCA Analysis Results

Replace text with Insert C

~~For the AP1000 large-break LOCA analysis, the best-estimate LOCA analysis methodology documented in Reference 34 is applied. The AP1000 large-break LOCA analysis complies with the restrictions in Reference 3 and Reference 32.~~ AP1000 sensitivity calculations evaluated the sensitivity to the modeling of the CMT and PRHR relative to the reference transient configuration. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the reference transient configuration. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was 2°F higher than the PCT of the reference transient configuration. The ASTRUM methodology samples the parameters ranged in the global model matrix of calculations, and the final 95 percent uncertainty calculations have been performed for AP1000. ~~Further, local and core-wide cladding oxidation values have been determined using the methodology approved in Reference 32.~~

Replace text with Insert D

Replace text with Insert E

~~In the AP1000 ASTRUM analysis, the same uncertainty calculation was the limiting PCT and maximum local oxidation (MLO) case. The limiting PCT/MLO case in the AP1000 ASTRUM analysis was a double ended guillotine split break.~~ Figures 15.6.5.4A-1 through 15.6.5.4A-12 present the parameters of principal interest for the limiting PCT~~/MLO~~ case. Values of the following parameters are presented:

- Highest calculated cladding temperature at any elevation for the five fuel rods modeled

- Hot rod cladding temperature transient at the limiting elevation for PCT
- Core fluid mass flows at the top of the core for the fuel assemblies modeled in WCOBRA/TRAC
- Pressurizer pressure
- Break flow rates
- Core and downcomer collapsed liquid levels
- Accumulator water flow rates
- Core makeup tank flow rates

15.6.5.4A.6 Description of AP1000 Large-Break LOCA Transient

A description of the limiting PCT~~/MLO~~ case from the AP1000 ASTRUM analysis follows. The limiting PCT~~/MLO~~ case is a double ended guillotine ~~split~~ break. The sequence of events is presented in Table 15.6.5-6. The break was modeled to occur in one of the cold legs in the loop containing the core makeup tanks. After the break opens, the vessel rapidly depressurizes and the core flow quickly reverses. The hot assembly fuel rods dry out and begin to heat up (Figures 15.6.5.4A-1 and 15.6.5.4A-2) after the initial flow reversal (Figure 15.6.5.4A-3).

remove comma

In Figure 15.6.5.4A-1, “Hot Rod” refers to the ~~hot fuel rod~~ at the maximum linear heat rate for the run, “Hot Assembly” refers to~~,~~ the average fuel rod in the hot assembly that contains the hot rod, “Support Column/Open Hole” refers to the fuel rod in average assemblies under support columns or open holes, “Guide Tubes” refers to the fuel rod in average assemblies under guide tubes, and “Low Power” refers to the fuel rod in the low power peripheral fuel assemblies.

The steam generator secondaries are assumed to be isolated immediately at the inception of the break, which maximizes their stored energy. The massive size of the break causes an immediate, rapid pressurization of the containment. At 2.2 seconds, an “S” signal is generated due to High-2 containment pressure. Applying the pertinent signal processing delay means that the valves isolating the core makeup tanks from the direct vessel injection line and the PRHR begin to open at 4.2 seconds into the transient. The reactor coolant pumps automatically trip after a 5.3 second delay from the actuation of the core makeup tank isolation valves, which is 9.5 seconds into the transient. Core shutdown occurs due to voiding; no credit is taken for the control rod insertion effect.

The system depressurizes rapidly (Figure 15.6.5.4A-4) as the initial mass inventory is depleted due to break flow. The pressurizer drains completely approximately 30 seconds into the transient, and accumulator injection commences 13 seconds into the transient (Figure 15.6.5.4A-5). Accumulator actuation shuts off core makeup tank flow (Figure 15.6.5.4A-6), which has been occurring since the isolation valve opened. The CMT liquid level remains well above the ADS Stage 1 actuation setpoint throughout the AP1000 LBLOCA cladding temperature excursion, even though CMT injection begins again around ~~190~~ seconds.

200

insert space

The dynamics of the 95th percentile estimator PCT~~/MLO~~ case are shown in terms of the flow rates of liquid, vapor, and entrained liquid at the top of the core (Figures 15.6.5.4A-7 through15.6.5.4A-9) for the peripheral, open hole/support column average power interior, and guide tube average power interior assemblies (the corresponding figure for the hot assembly is Figure 15.6.5.4A-3).

20 Insert F

Figure 15.6.5.4A-7 demonstrates that liquid downflow exists through the top of the peripheral core assemblies from approximately 1 to 3 seconds and again from 9 to ~~21~~ seconds in the 95th percentile estimator PCT~~/MLO~~ case. The power of the fuel in this region is significantly lower than that of the fuel in the open hole/support column and guide tube locations (Table 15.6.5-4), so liquid downflow occurs earlier on the periphery than in the average power assemblies.

is delayed relative to downflow into the guide tubes; there is continuous liquid flow from approximately 10 seconds

28

Figure 15.6.5.4A-9 presents the open hole/support column assembly top of core flow behavior. In this case, ~~there is~~ liquid downflow into the support column/open hole assemblies ~~from 11 seconds~~ until 22 seconds; the entrained liquid flow continues to be significant until ~~30~~ seconds as fluid drains through the upper core plate holes into the upper plenum. ~~Figure 15.6.5.4A-8 shows that in the 95th percent estimator PCT/MLO case, the liquid downflow into the guide tube assemblies occurs from 14 seconds to 22 seconds; in this case the downflow is delayed relative to the liquid downflow in the support column/open hole assemblies due to the steam generation and resulting significant vapor upflow at the top of active fuel in the guide tube assemblies during blowdown.~~

Around 10

The timing of the initial downflow into the hot assembly is similar to that of the downflow into the open hole/support column average assemblies. ~~By 11~~ seconds into the transient, liquid that has built up in the global region above the hot assembly begins to flow into the hot assembly (Figure 15.6.5.4A-3). Significant flow of continuous liquid into the hot assembly exists between ~~11 to 22~~ seconds. The liquid flow is not enough to quench the hot rod and hot assembly rod or the average rods at all elevations (Figure 15.6.5.4A-1) although some cooling is achieved.

10 to 20

27.5 40.0

After 13 seconds into the transient, the accumulator begins to inject water into the upper downcomer region, most of which is initially bypassed to the break. The break flow rate diminishes as the transient progresses (Figure 15.6.5.4A-10). At ~~28.5~~ seconds, ~~the accumulator~~ injection begins to refill the lower plenum. At approximately ~~41.0~~ seconds, the lower plenum fills to the point that water begins to reflood the core from below (Figure 15.6.5.4A-11). The void fraction at the core bottom begins to decrease, and as time passes, core cooling increases substantially. Figure 15.6.5.4A-11 presents the collapsed liquid levels in the core; Figure 15.6.5.4A-12 presents the collapsed liquid levels in the downcomer. The cladding temperature begins to decrease once the core water level has risen high enough in the core.

15.6.5.4A.7 Global Model Sensitivity Studies and Uncertainty Evaluation

Section 15.6.5.4A discusses the treatment of the global model parameters and the uncertainty evaluation in the ASTRUM methodology.

15.6.5.4A.8 Large-Break LOCA Conclusions

In accordance with 10 CFR 50.46, the conclusions of the best-estimate large-break LOCA analysis are that there is a high level probability that the following criteria are met.

1. The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature (PCT)) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (i.e., maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

   Note that criterion 4 has historically been satisfied by adherence to criteria 1 and 2, and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. Criteria 1 and 2 are satisfied for best-estimate large-break LOCA applications. The approved methodology specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low power channel as defined in the WCOBRA/TRAC model. This situation has not been calculated to occur for the AP1000. Therefore, acceptance criterion 4 is satisfied.

5. After successful initial operation of the emergency core cooling system (ECCS), the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

   Criterion 5 is satisfied if a coolable core geometry is maintained and the core is cooled continuously following the LOCA. The AP1000 passive core cooling system provides effective core cooling following a large-break LOCA event, even assuming the limiting single failure of a core makeup tank delivery line isolation valve. The large-break LOCA

to

transient has been extended beyond fuel rod quench the time at which the CMT liquid level has decreased to the ~~low-2~~ setpoint that actuates the fourth-stage ADS valves and IRWST injection. A significant increase in safety injection flow rate occurs when the IRWST becomes active. The analysis performed demonstrates that CMT injection is sufficient to maintain the mass inventory in the core and downcomer, from the period of fuel rod quench until IRWST injection. The AP1000 passive core cooling system provides effective post-LOCA long-term core cooling (Section 15.6.5.4C).

when the burnup-related effects of thermal conductivity degradation and peaking factor burndown are considered.

Table 15.6.5-8 presents the calculated 95th percentile PCT, maximum cladding oxidation, maximum hydrogen generation, and core cooling results.

Based on the analysis, the Westinghouse Best-Estimate Large-Break LOCA methodology has shown that the acceptance criteria of 10 CFR 50.46 are satisfied for AP1000~~.~~

15.6.5.4B Small-break LOCA Analyses

Should a small break LOCA occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal occurs when the pressurizer low-pressure trip setpoint is reached. An "S" signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

- Reactor trip leads to a rapid reduction of power to a residual level corresponding to fission product decay heat by the insertion of control rods to shut down the reactor.
- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

15.6.5.4B.1 Description of Small-break LOCA Transient

The AP1000 plant design includes passive safety features to prevent or minimize core uncovery during small-break LOCAs. The passive safety design approach of the AP1000 is to depressurize the reactor coolant system if the break or leak is greater than the capability of the makeup system or if the non-safety makeup system fails to perform. By depressurizing the reactor system, large volumes of borated water in the accumulators and in the IRWST become available for cooling the core. These analyses demonstrate that, with a single failure, the passive systems are capable of depressurizing the reactor coolant system while maintaining acceptable core conditions and establishing stable delivery of cooling water from the IRWST.

During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurizer low-pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned for delivery following the generation of an "S" signal when the pressurizer low-pressure setpoint

At the time recirculation begins, the containment level will be about 109.3 feet (for a non-DVI LOCA) and will be about 108.0 feet (for a DVI LOCA). Over a period of weeks after a LOCA, water may slowly leak from the flooded areas in containment to other areas inside containment that did not initially flood. As a result, the minimum containment water could decrease to 103.5 feet. During recirculation operation following a LOCA and ADS actuation, the operators are guided to maintain the containment water level above the 107-foot elevation by adding borated water to the containment. In addition, if the plant continues to operate in the recirculation mode, the operators are guided to increase the level to 109 feet within 30 days of the accident. These actions provide additional margin in water flow through the ADS Stage 4 line. The operators are also guided to sample the hot leg boron concentration prior to initiating recovery actions that might introduce low temperature water to the reactor.

15.6.5.4C.5 Conclusions

Calculations of AP1000 long-term cooling performance have been performed using the WCOBRA/TRAC model developed for AP1000 and described in Reference 24. The DEDVI case was chosen because it reaches sump recirculation at the earliest time (and highest decay heat). A window mode case at the minimum containment water level postulated to occur 2 weeks into long-term cooling was also performed.

The DEDVI small-break LOCA exhibits no core uncovery due to its adequate reactor coolant system mass inventory condition during the long-term cooling phase from initiation into containment recirculation. Adequate flow through the core is provided to maintain a low cladding temperature and to prevent any buildup of boric acid on the fuel rods. The wall-to-wall floodup case using the window mode technique demonstrates that effective core cooling is also provided at the minimum containment water level. The results of these cases demonstrate the capability of the AP1000 passive systems to provide long-term cooling for a limiting LOCA event.

15.6.6 References

1. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," and Appendix K to 10 CFR 50, "ECCS Evaluation Models."

2. American Nuclear Society Proposed Standard, ANS 5.1 "Decay Energy Release Rates Following Shutdown of Uranium-Cooled Thermal Reactors," October (1971), Revised October (1973).

3. "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," NUREG-1512, September 1998.

4. Not used.

5. "Emergency Core Cooling Systems; Revision to Acceptance Criteria," Federal Register, Vol. 53, No. 180, September 16, 1988.

6. Not used.

7. "AP600 Design Control Document," Revision 3, December 1999.

8. Letter from R. C. Jones, Jr., (USNRC), to N. J. Liparulo, (W), Subject: "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse CQD for Best Estimate LOCA Analysis," June 28, 1996.

9. Not used.

10. Bajorek, S. M., et al., "Code Qualification Document for Best-Estimate LOCA Analysis," WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, and WCAP-14747 (Non-Proprietary), 1998.

11. Hochreiter, L. E., et al., "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident," WCAP-14171, Revision 2 (Proprietary) and WCAP-14172, Revision 2 (Nonproprietary), March 1998.

12. Meyer, P. E., "NOTRUMP - A Nodal Transient Small-Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Nonproprietary), August 1985.

13. Lee, N., Rupprecht, S. D., Schwarz, W. R., and Tauche, W. D., "Westinghouse Small-Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Nonproprietary), August 1985.

14. Carlin, E. L., Bachrach, U., "LOFTRAN & LOFTTR2 AP600 Code Applicability Document," WCAP-14234, Revision 1 (Proprietary) and WCAP-14235, Revision 1 (Nonproprietary), August 1997.

15. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

16. Friedland, A. J., Ray S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

17. Kemper, R. M., "AP600 Accident Analyses B Evaluation Models," WCAP-14601, Revision 2 (Proprietary) and WCAP-15062, Revision 2 (Nonproprietary), May 1998.

18. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908-A, December 1989.

19. Soffer, L., et al., NUREG-1465, "Accident source Terms for Light-Water Nuclear Power Plants," February 1995.

20. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

21. Lewis, R. N., et al., "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10698-P-A (Proprietary) and WCAP-10750-A (Nonproprietary), August 1987.

22. "NOTRUMP Final Validation Report for AP600," WCAP-14807, Revision 5 (Proprietary) and WCAP-14808, Revision 2 (Nonproprietary), August 1998.

23. Garner, D. C., et al., "WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report," WCAP-14776, Revision 4 (Proprietary) and WCAP-14777, Revision 4 (Nonproprietary), April 1998.

24. "AP1000 Code Applicability Report," WCAP-15644-P (Proprietary) and WCAP-15644-NP (Nonproprietary), Revision 2, March 2004.

25. "AP1000 PIRT and Scaling Assessment," WCAP-15613 (Proprietary) and WCAP-15706 (Nonproprietary), February 2001.

26. Kemper, R. M., "Applicability of the NOTRUMP Computer Code to AP600 SSAR Small-Break LOCA Analyses," WCAP-14206 (Proprietary) and WCAP-14207 (Nonproprietary), November 1994.

27. Not used.

28. Zuber, et al., "The Hydrodynamic Crisis in Pool Boiling of Saturated and Subcooled Liquids," Part II, No. 27, International Developments in Heat Transfer, 1961.

29. Griffith, et al., "PWR Blowdown Heat Transfer," Thermal and Hydraulic Aspects of Nuclear Reactor Safety, ASME, New York, Volume 1, 1977.

30. Chang, S. H. et al. "A study of critical heat flux for low flow of water in vertical round tubes under low pressure," Nuclear Engineering and Design, 132, 225-237, 1991.

31. ~~Not used.~~ LTR-NRC-12-86, "Westinghouse Response to NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report' (Proprietary/ Non-Proprietary)," Westinghouse Electric Company LLC.

32. Nissley, M. E., et al., 2005, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P-A and WCAP-16009-NP-A (Non-proprietary).

33. Dederer, S. I., et al., 1999, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," WCAP-14449-P-A, Revision 1 and WCAP-14450 (Non-proprietary).

34. APP-GW-GLE-026, Revision 1, 2009, "Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000," Westinghouse Electric Company LLC.

35. "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design," NUREG-1793, Supplement 2, August 2011.

Table 15.6.5-4

MAJOR PLANT PARAMETER ASSUMPTIONS USED IN THE BEST-ESTIMATE LARGE-BREAK LOCA ANALYSIS

Parameter	Value
Plant Physical Configuration	
Steam generator tube plugging level	≤ 10% (10% tube plugging bounds 0%)
Hot assembly location	Under support column (Bounds under open hole or guide tube)
Pressurizer location	In intact loop (Bounds location in broken loop)
Initial Operating Conditions	
Reactor power	Core power < 1.01*3400 MWt
Peak linear heat rate	$F_Q \leq 2.6$ See Table 15.6.5-7
Hot rod assembly power	$F_{\Delta H} \leq 1.72$ See Table 15.6.5-7
Hot assembly power	$P_{HA} \leq 1.654$
Axial power distribution[1]	See Figure 15.6.4A-13
Peripheral assembly power	$0.2 \leq P_{LOW} \leq 0.8$
Fluid Conditions	
Reactor coolant system average temperature	$573.6 - 8.0°F \leq T_{AVG} \leq 573.6 + 8.0°F$
Pressurizer pressure	2250 ± 50 psia
Pressurizer level (water volume)	1000 ft^3 (nominal)
Accumulator temperature	$50°F \leq T_{ACC} \leq 120°F$
Accumulator pressure	652 psia ≤ P_{ACC} ≤ 784 psia
Accumulator water volume	$1666.8\ ft^3 \leq V_{ACC} \leq 1732.3\ ft^3$
Reactor Coolant System Boundary Conditions	
Single failure assumption	Failure of one CMT isolation valve to open
Offsite power availability	Available (Bounds loss of offsite power at time zero)
Reactor coolant pump automatic trip delay time after receiving S-signal	5.3 s
Containment pressure	Bounded (minimum)

Note:

1. Treatment of axial power distribution consistent with WCAP-16009-P-A (Reference 32) methodology.

Table 15.6.5-5

AP1000 LOCA CHRONOLOGY

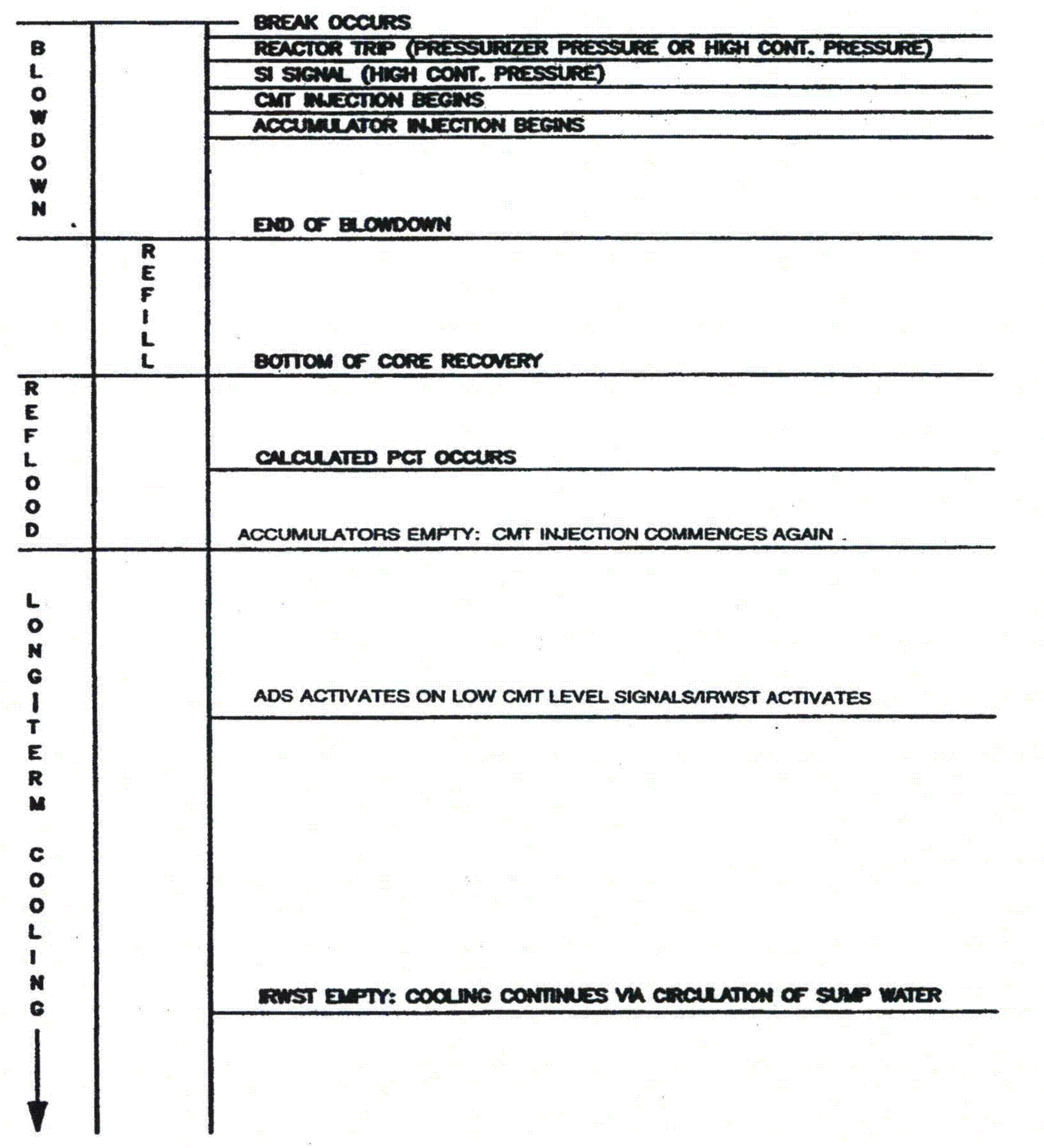

Table 15.6.5-6

BEST-ESTIMATE LARGE-BREAK SEQUENCE OF EVENTS FOR THE LIMITING PCT~~/MLO~~ CASE

Event	Time (seconds)
Break initiation	0.0
Safeguards signal	2.2
CMT isolation valves begin to open	4.2
Reactor coolant pumps trip	9.5
Accumulator injection begins	~13
End of blowdown	~~28.5~~ 27.5
Bottom of core recovery	39.5 ~~41~~
Calculated PCT occurs	~~~60~~ ~58
Core quench occurs	~240 ~~160~~
CMT injection resumes	~~~190~~ ~200
End of transient	265 ~~255~~

$\boxed{\text{Insert Table 15.6.5-7}}$

~~Table 15.6.5-7 is Not Used~~

Table 15.6.5-8

BEST-ESTIMATE LARGE-BREAK LOCA RESULTS

10 CFR 50.46 Requirement	Value	Criteria
Calculated 95th percentile PCT (°F)	~~1867~~* 1936	≤ 2200
Maximum local cladding oxidation (%)	~~3.10~~ 4.2	≤ 17
Maximum core-wide cladding oxidation (%)	~~0.27~~ 0.30	≤ 1
Coolable geometry	Core remains coolable	Core remains coolable
Long-term cooling	Core remains cool in long term	Core remains cool in long term

*Value contains 2°F bias for PCT sensitivity to PRHR isolation

Replace with updated figure

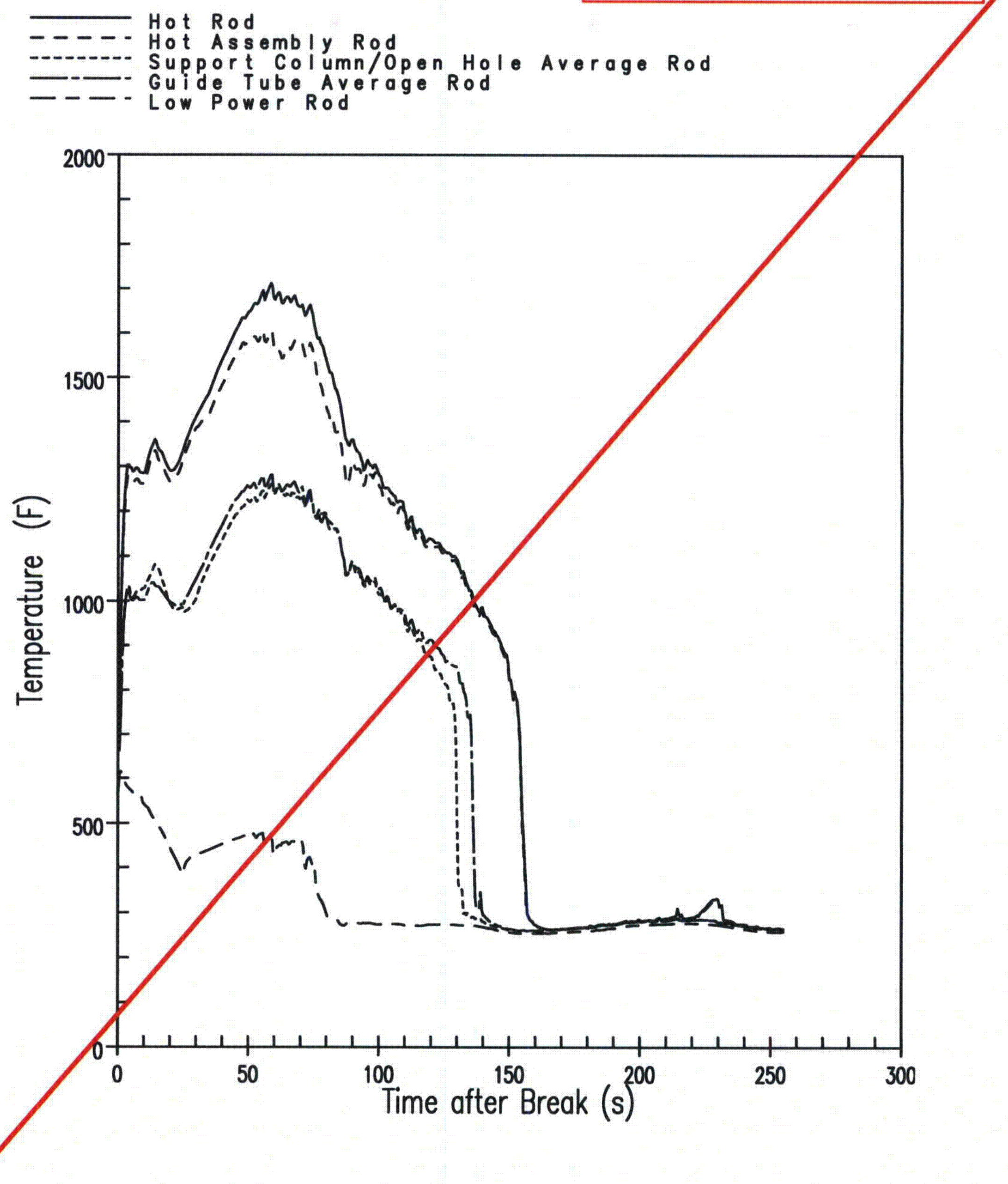

Figure 15.6.5.4A-1

WCOBRA/TRAC Peak Cladding Temperature for All Five Rod Groups for 95th Percentile Estimator PCT/~~MLO~~ Case

Replace with updated figure

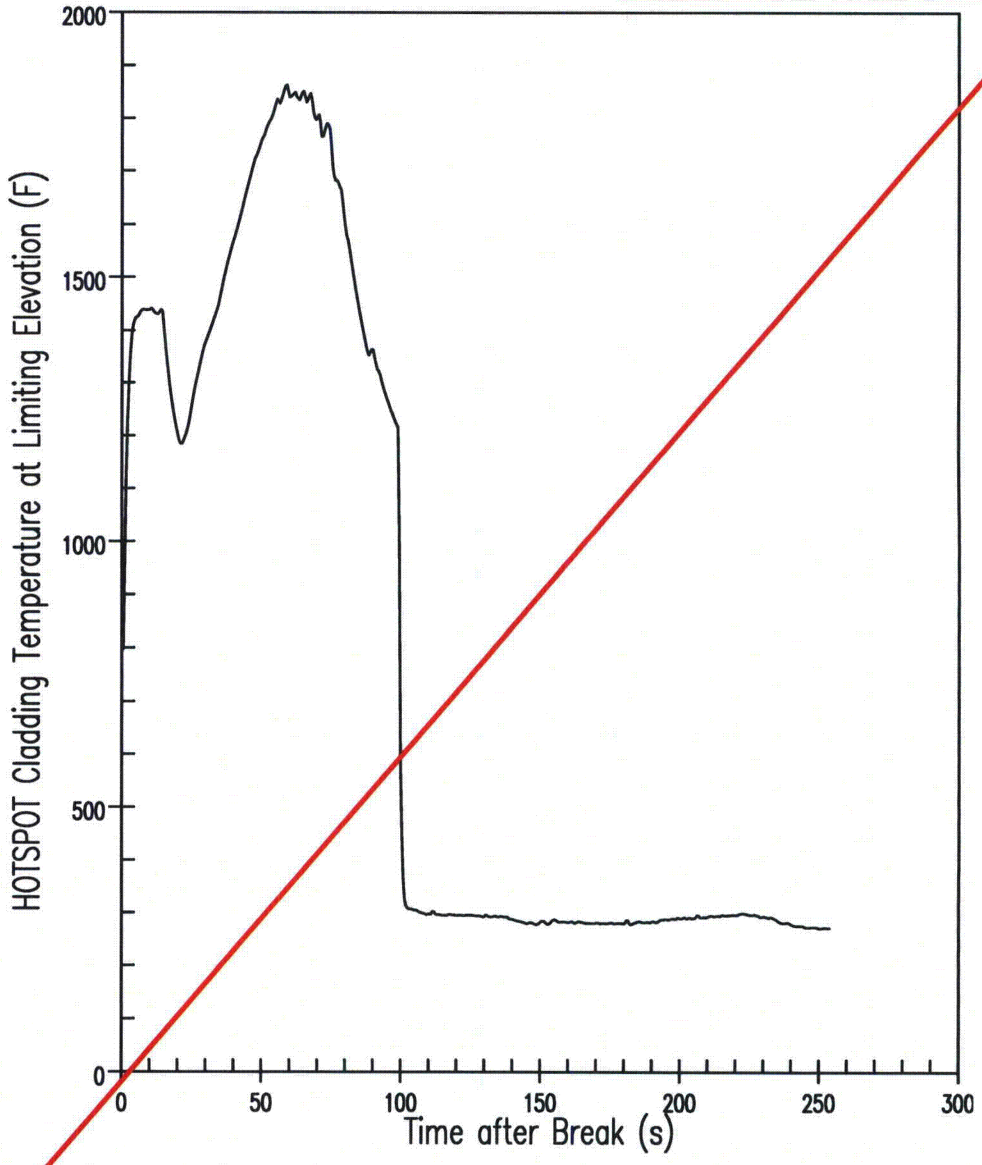

Figure 15.6.5.4A-2

HOTSPOT Cladding Temperature Transient at Limiting Elevation for 95th Percentile Estimator PCT/~~MLO~~ Case

Replace with updated figure

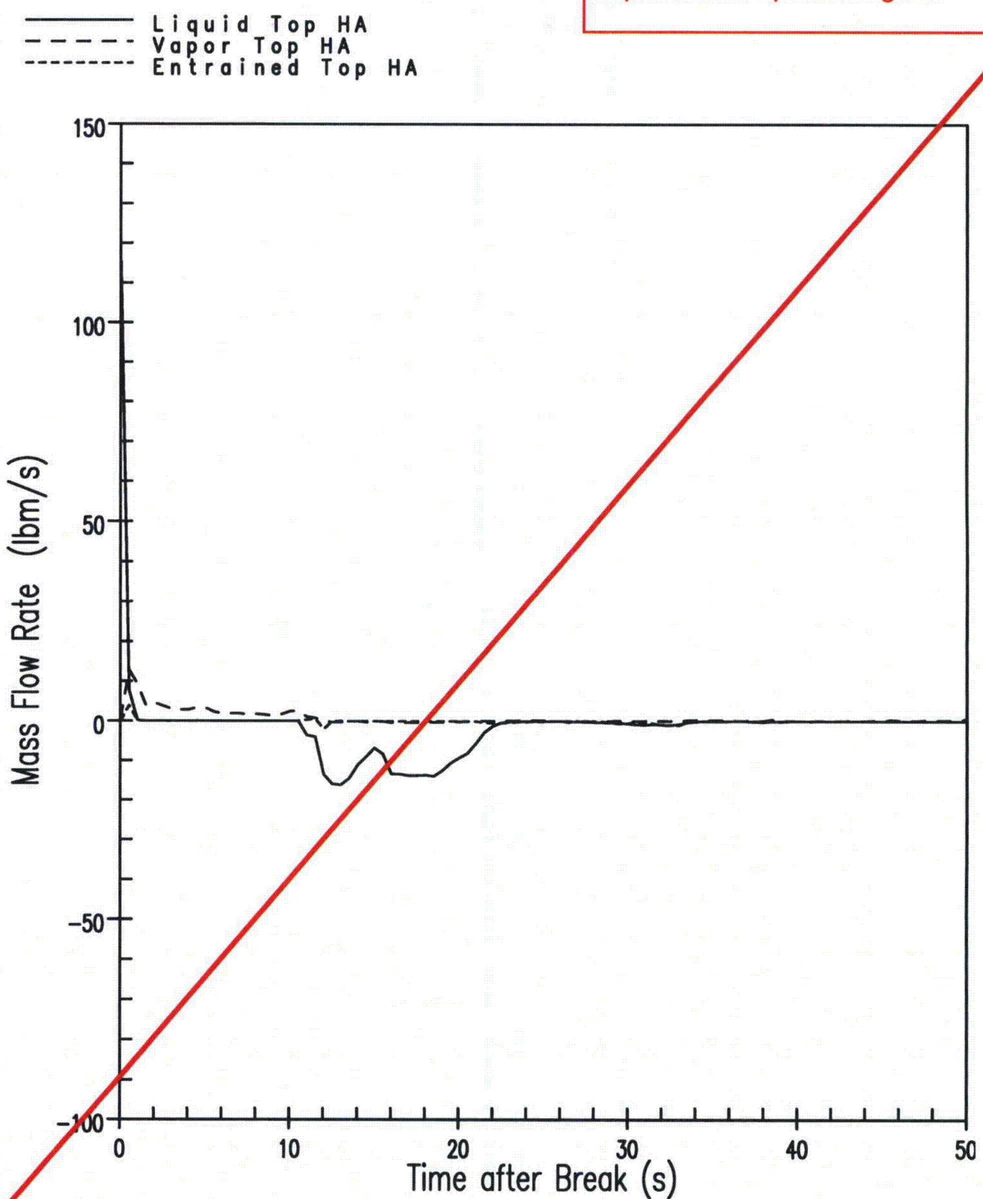

Figure 15.6.5.4A-3

Mass Flow at Top of Hot Assembly Channel for 95th Percentile Estimator PCT/~~MLO~~ Case

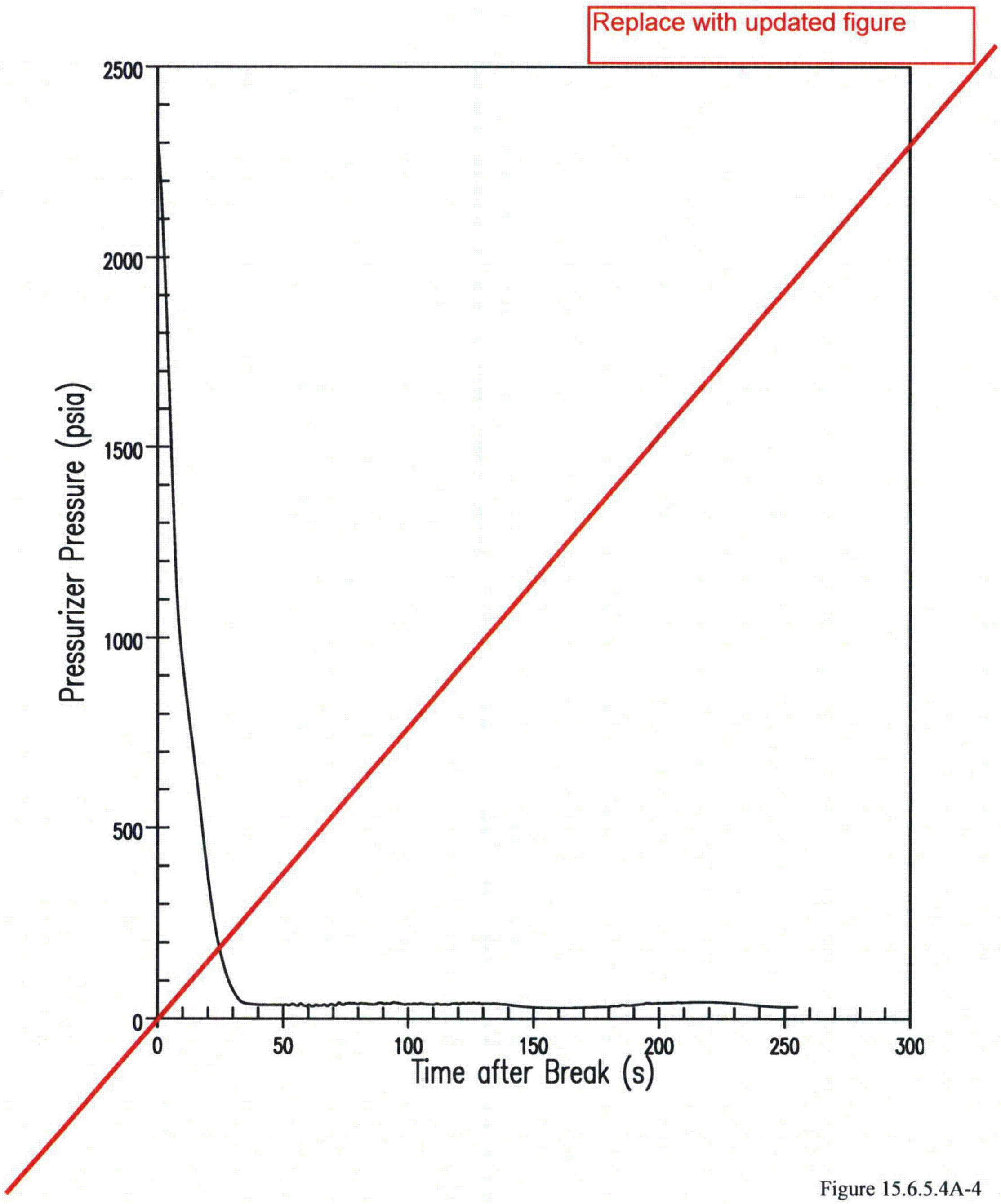

Figure 15.6.5.4A-4

Pressurizer Pressure for
95th Percentile Estimator PCT/~~MLO~~ Case

Replace with updated figure

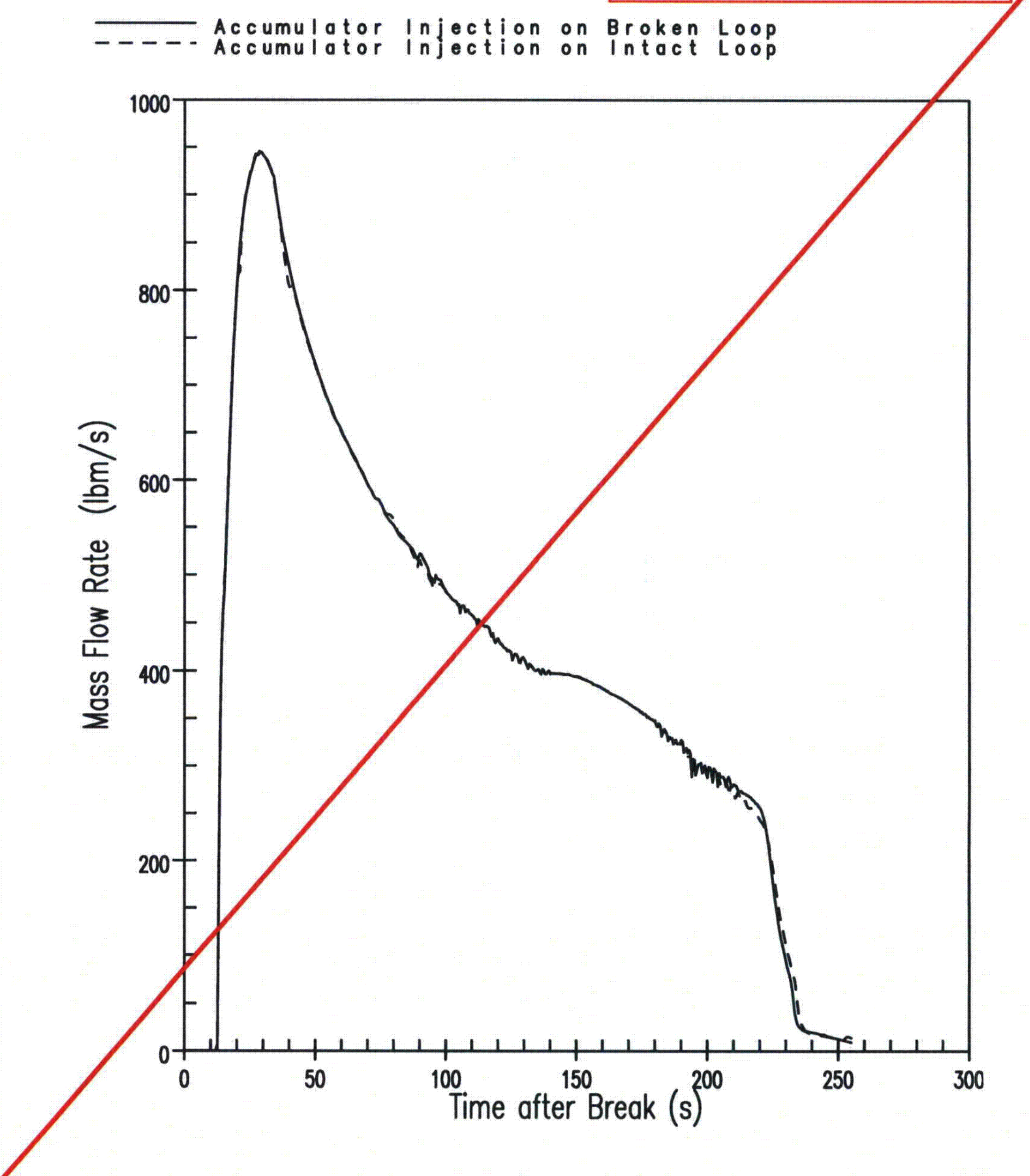

Figure 15.6.5.4A-5

Accumulator Injection Flow for
95th Percentile Estimator PCT/~~MLO~~ Case

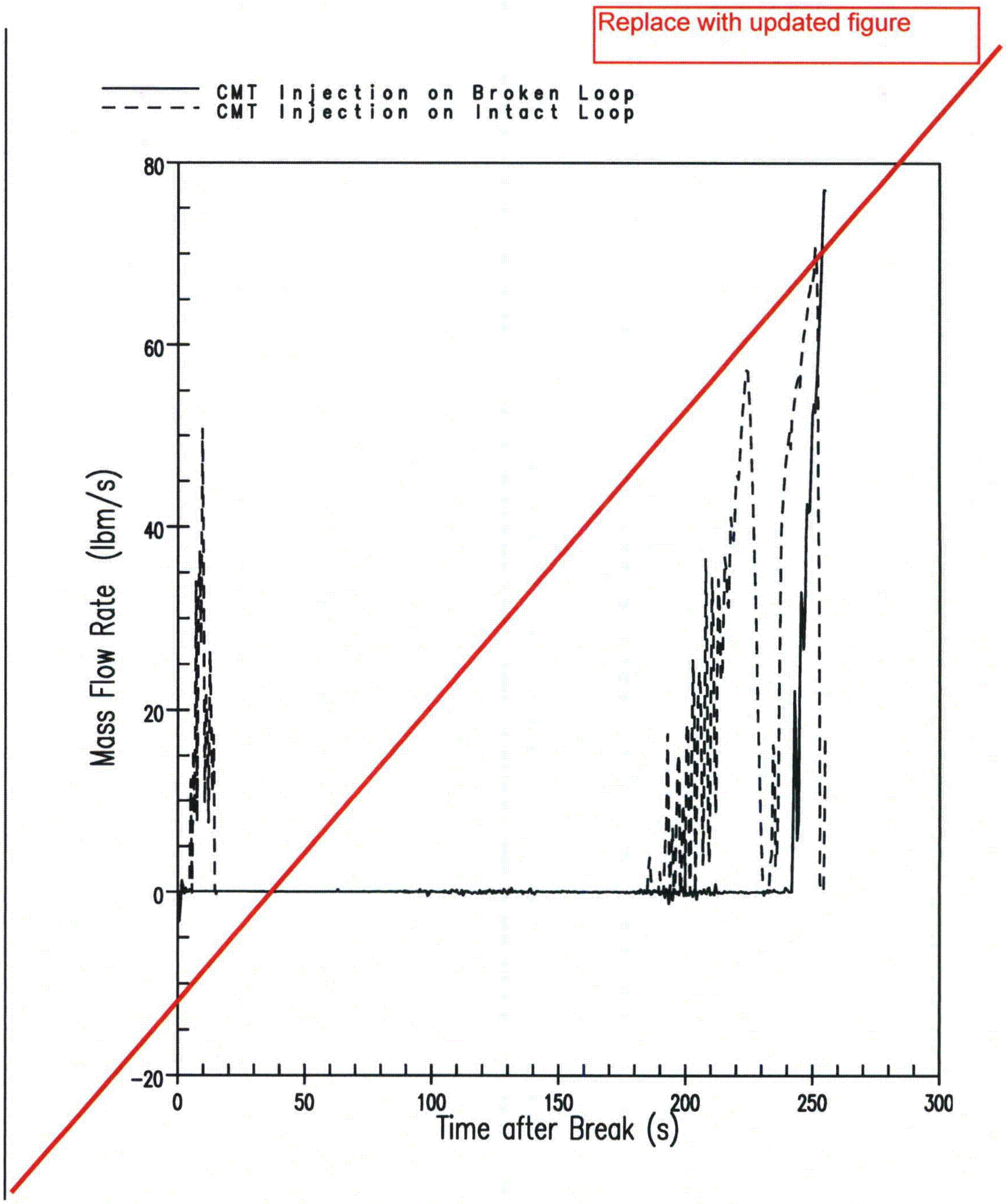

Figure 15.6.5.4A-6

Core Makeup Tank Injection Flow
for 95th Percentile Estimator PCT/~~MLO~~ Case

Replace with updated figure

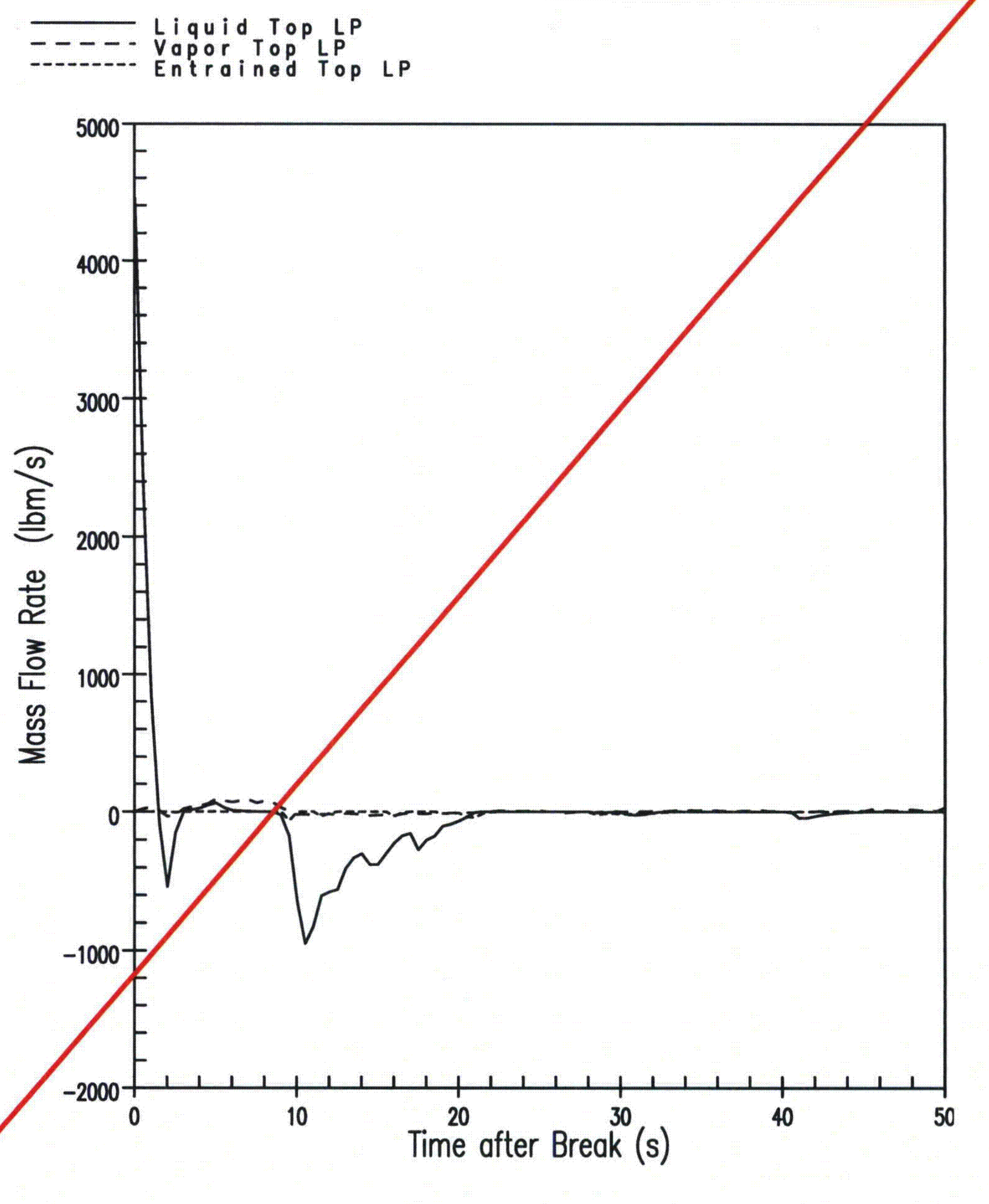

Figure 15.6.5.4A-7

Mass Flow at Top of Peripheral Assemblies
Channel for 95th Percentile Estimator PCT/~~MLO~~ Case

Replace with updated figure

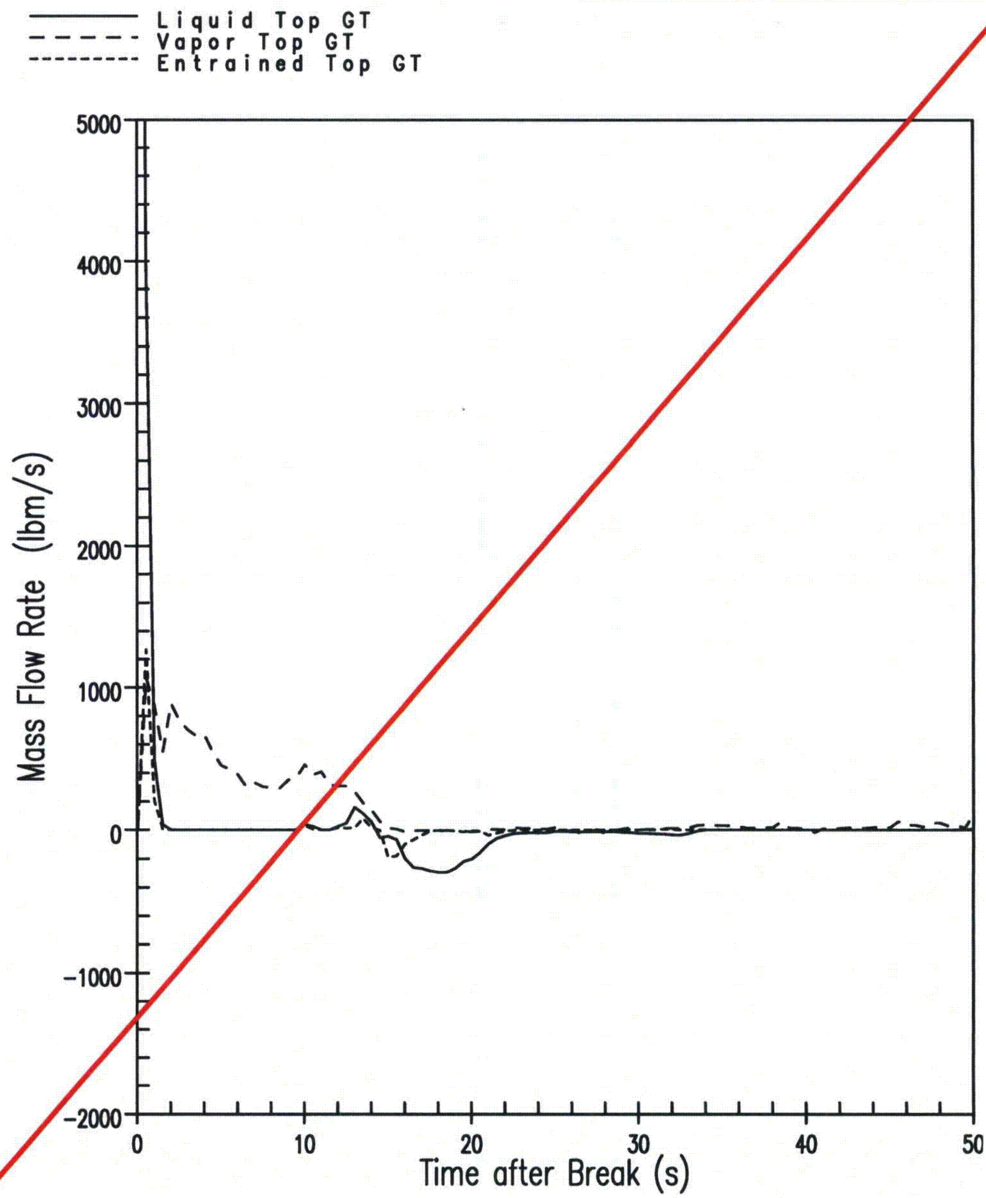

Figure 15.6.5.4A-8

Mass Flow at Top of Guide Tube Assemblies Channel for 95th Percentile Estimator PCT/~~MLO~~ Case

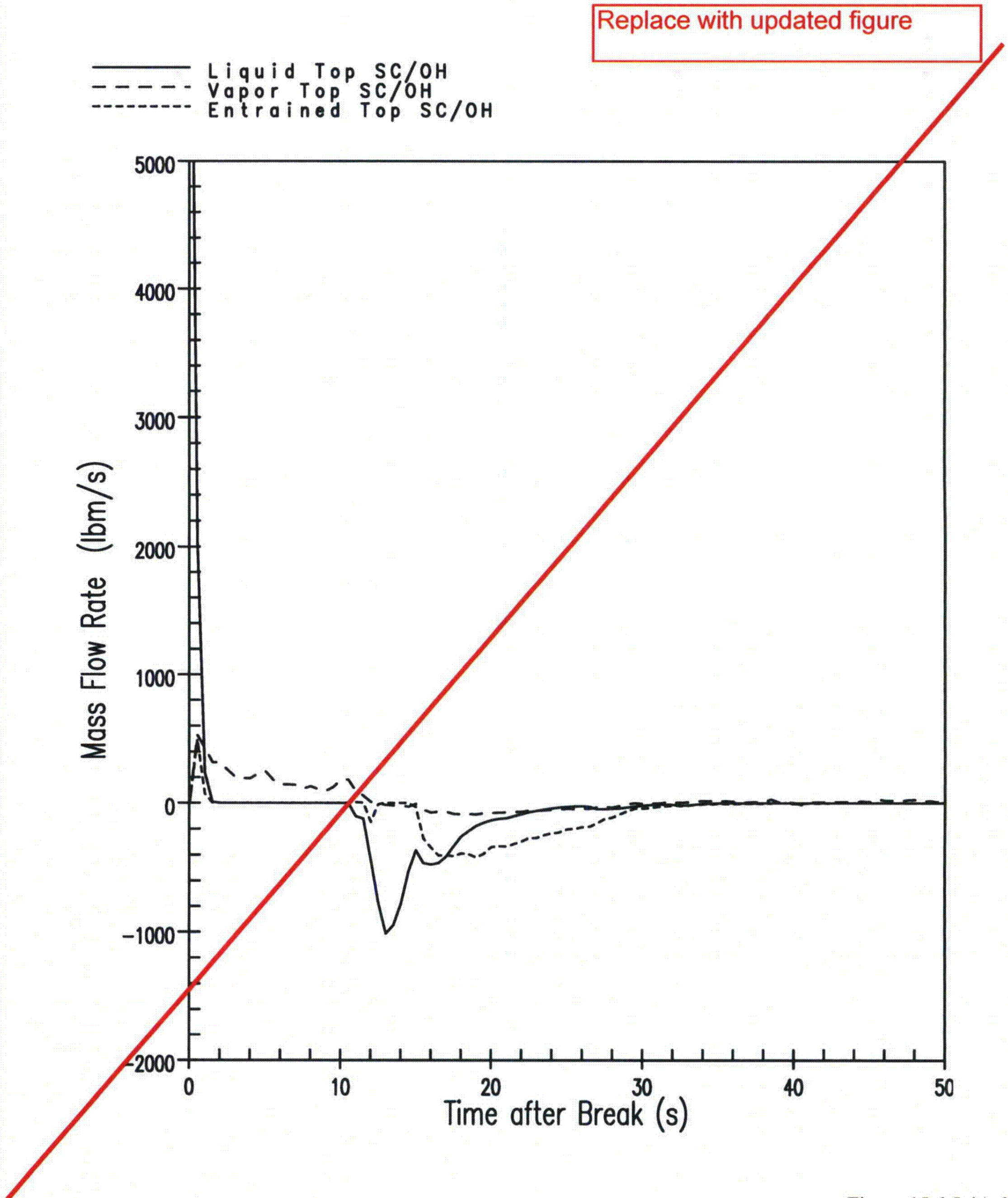

Figure 15.6.5.4A-9

Mass Flow at Top of Support Column/Open Hole Assemblies Channel for 95th Percentile Estimator PCT/~~MLO~~ Case

Replace with updated figure

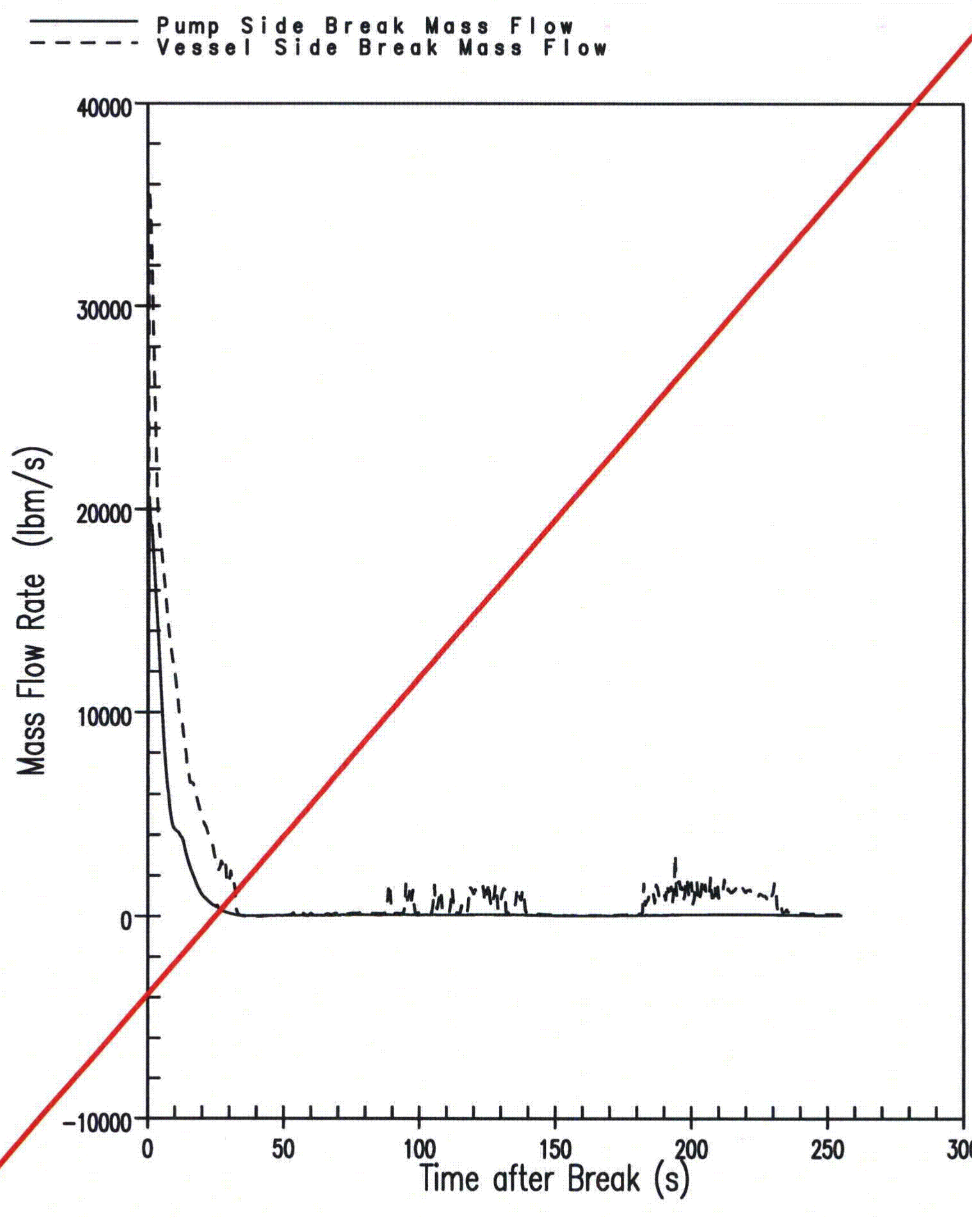

Figure 15.6.5.4A-10

Break Mass Flow for
95th Percentile Estimator PCT~~/MLO~~ Case

Replace with updated figure

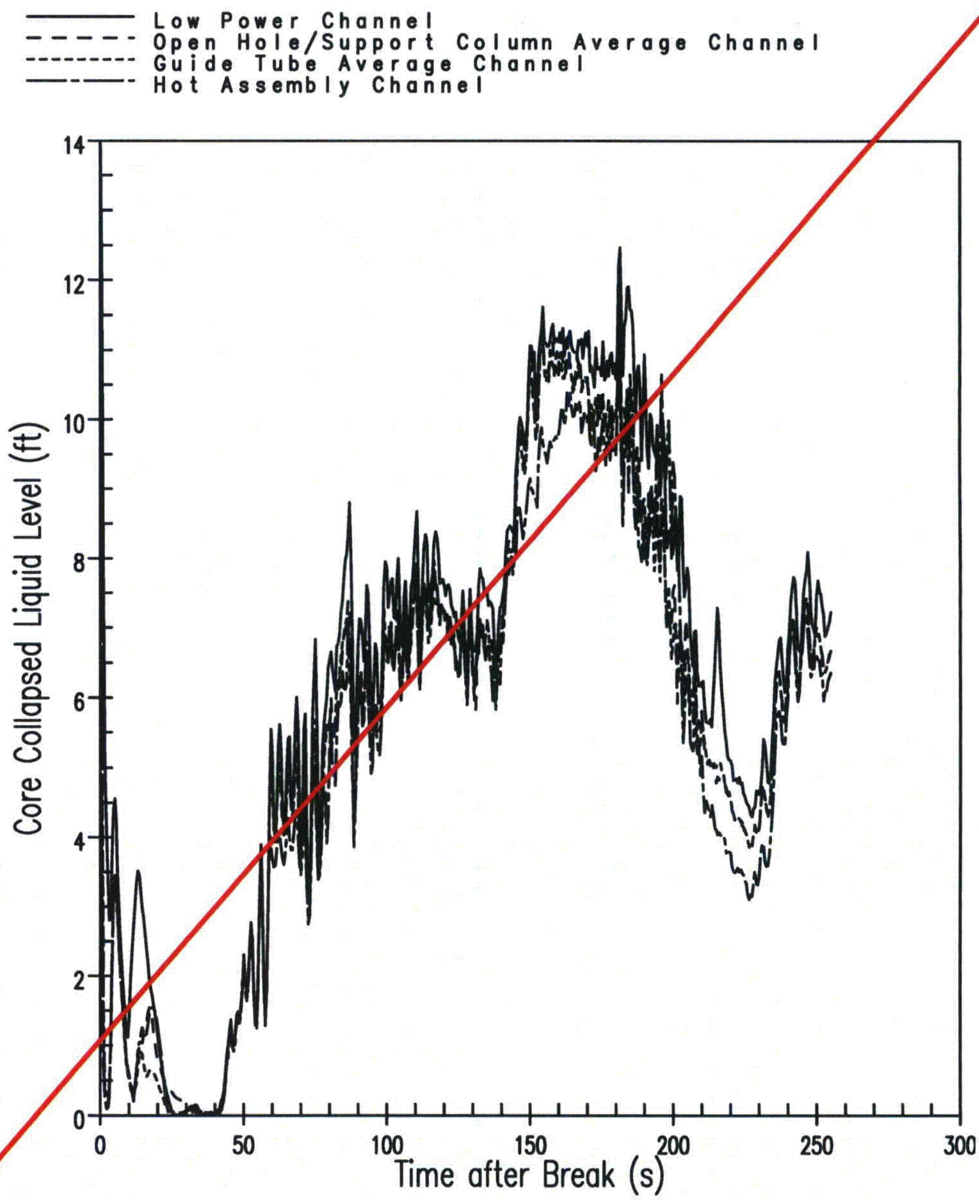

Figure 15.6.5.4A-11

Core Channel Collapsed Liquid Levels
for 95th Percentile Estimator PCT/~~MLO~~ Case
(Reference Point: Bottom of Active Fuel)

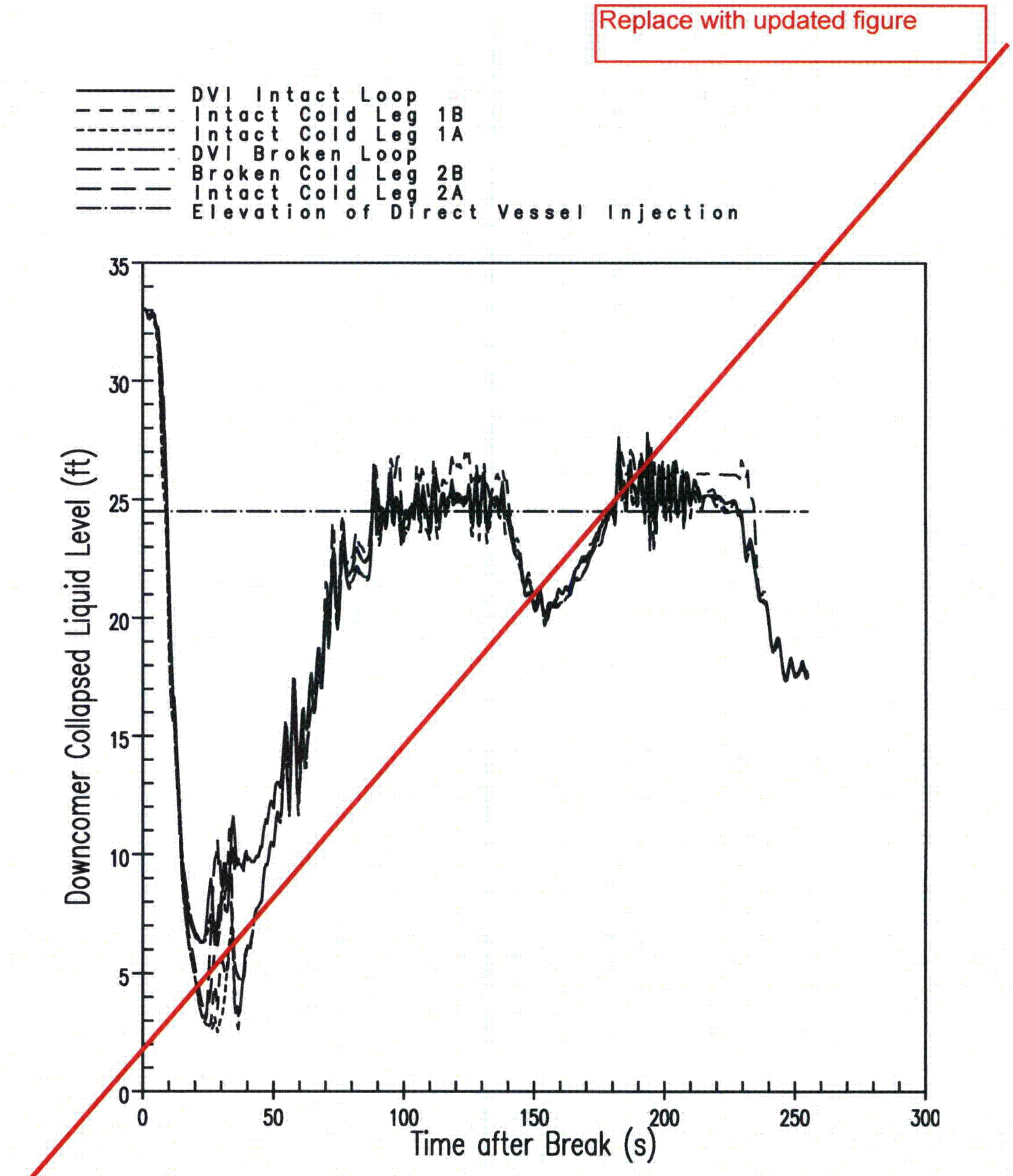

Figure 15.6.5.4A-12

Downcomer Channel Collapsed Liquid Levels for 95th Percentile Estimator PCT~~/MLO~~ Case (Reference Point: Inside Bottom of Vessel)

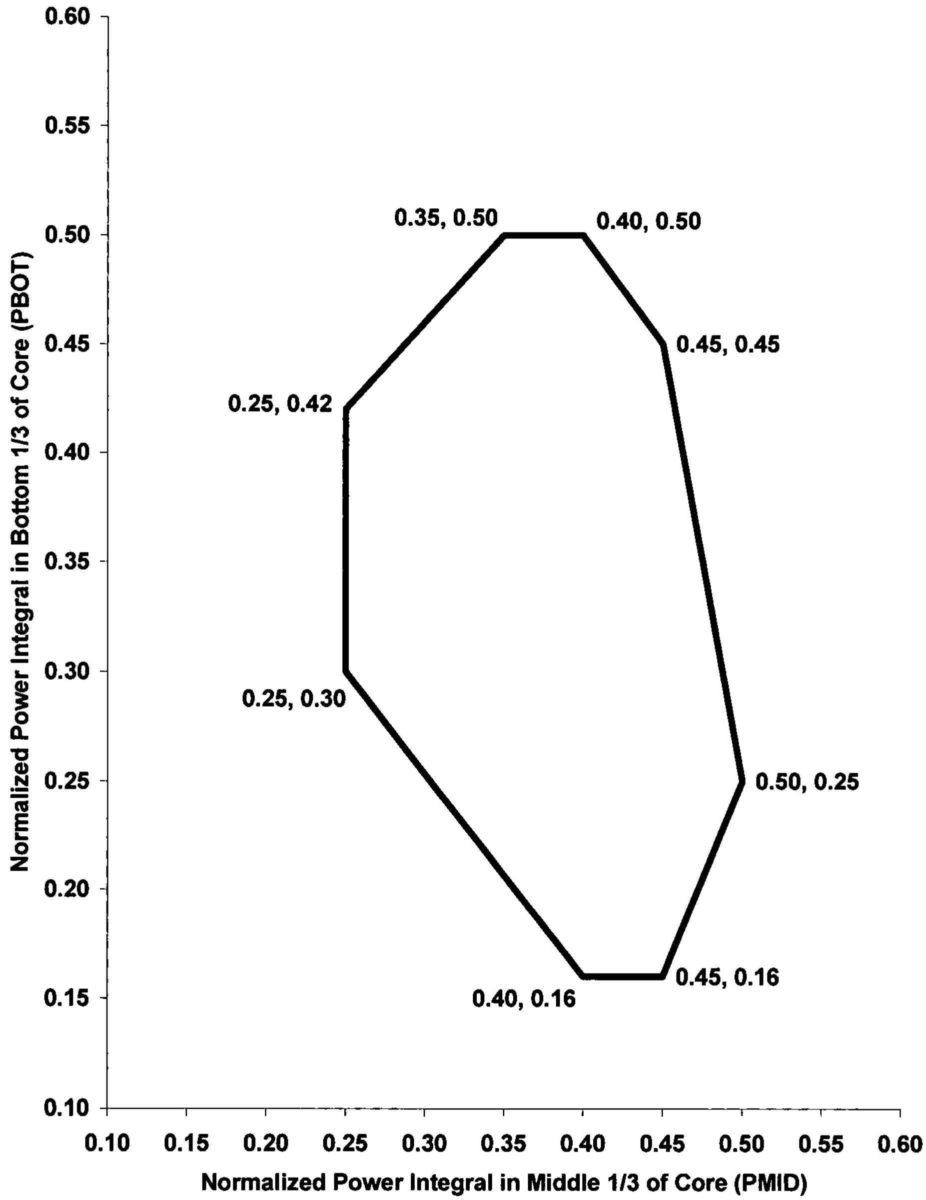

Figure 15.6.5.4A-13

PBOT/PMID Box Supported by AP1000 ASTRUM Analysis

CRR MARKUP TEXT INSERTS, UPDATED TABLE, UPDATED FIGURES

Insert A.
Application of the ASTRUM methodology for the AP1000 plant was submitted to the NRC staff per Reference 34. The NRC staff has reviewed and approved the ASTRUM methodology for estimating the 95th percentile PCT for the AP1000 plant, as documented in Reference 35. In the ASTRUM methodology,

Insert B.
For the AP1000 large-break LOCA analysis, a plant-specific adaptation of the ASTRUM methodology is applied as described in Reference 31. The plant-specific adaptation explicitly models the effects of thermal conductivity degradation and peaking factor burndown. The best-estimate large-break LOCA analysis complies with the stipulated applicability limits in the Reference 3, Reference 32, and Reference 35 approvals.

Insert C.
For the AP1000 large-break LOCA analysis, a plant-specific adaptation of the ASTRUM best-estimate LOCA analysis methodology is applied, as described in Reference 31. The AP1000 large-break LOCA analysis complies with the restrictions in Reference 3, Reference 32, and Reference 35.

Insert D.
Further, local and core-wide cladding oxidation values have been determined using the plant-specific adaptation of the approved Reference 32 methodology as described in Reference 31.

Insert E.
In the AP1000 ASTRUM analysis, the limiting PCT and limiting MLO results were from two different uncertainty calculations. Both the limiting PCT case and the limiting MLO case were double ended guillotine breaks.

Insert F.
Once the upper head begins to flash, liquid drains directly down the guide tubes and that fraction that is able to penetrate into the core does so, at a maximum flow rate exceeding 1000 lbm/sec of total liquid flow between 5-23 seconds (Figure 15.6.5.4A-8).

Table 15.6.5-7. Summary of Peaking Factor Burndown Supported by AP1000 Plant Best Estimate Large Break LOCA CRR Updated Analysis Considering TCD

Hot Rod Burnup (GWd/MTU)	FdH (includes uncertainties)(1)	FQ Transient (Max FQ, includes uncertainties)	FQ SS Baseload (without uncertainties)
0	1.72	2.60	2.10
30	1.72	2.60	2.10
49	1.55	2.30	1.85
55	1.55	2.30	1.85
62	1.40	1.90	1.45

(1) Hot assembly power follows the same burndown, since it is a function of FdH

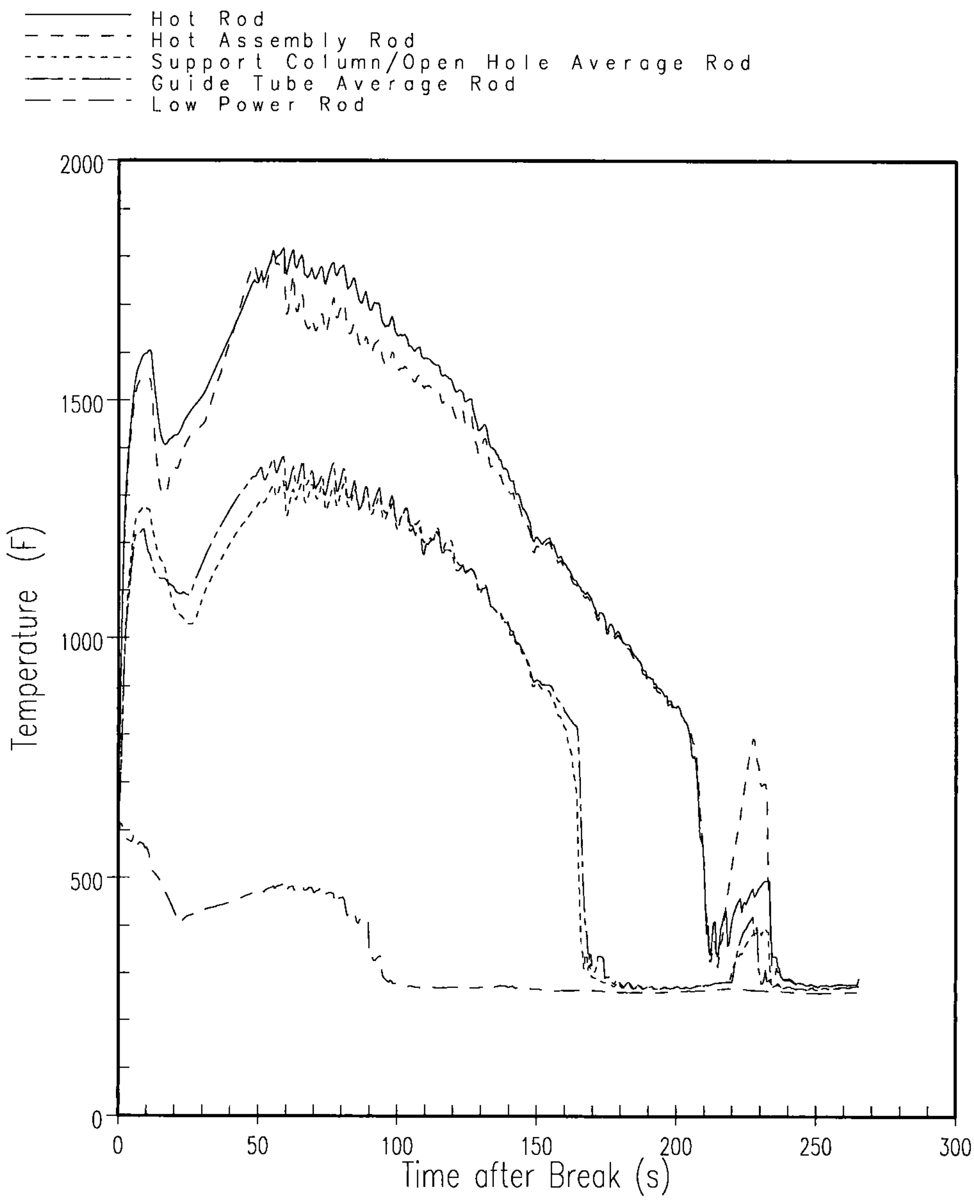

Figure 15.6.5.4A-1
WCOBRA/TRAC Peak Cladding Temperature for All Five Rod Groups for 95th Percentile Estimate PCT Case

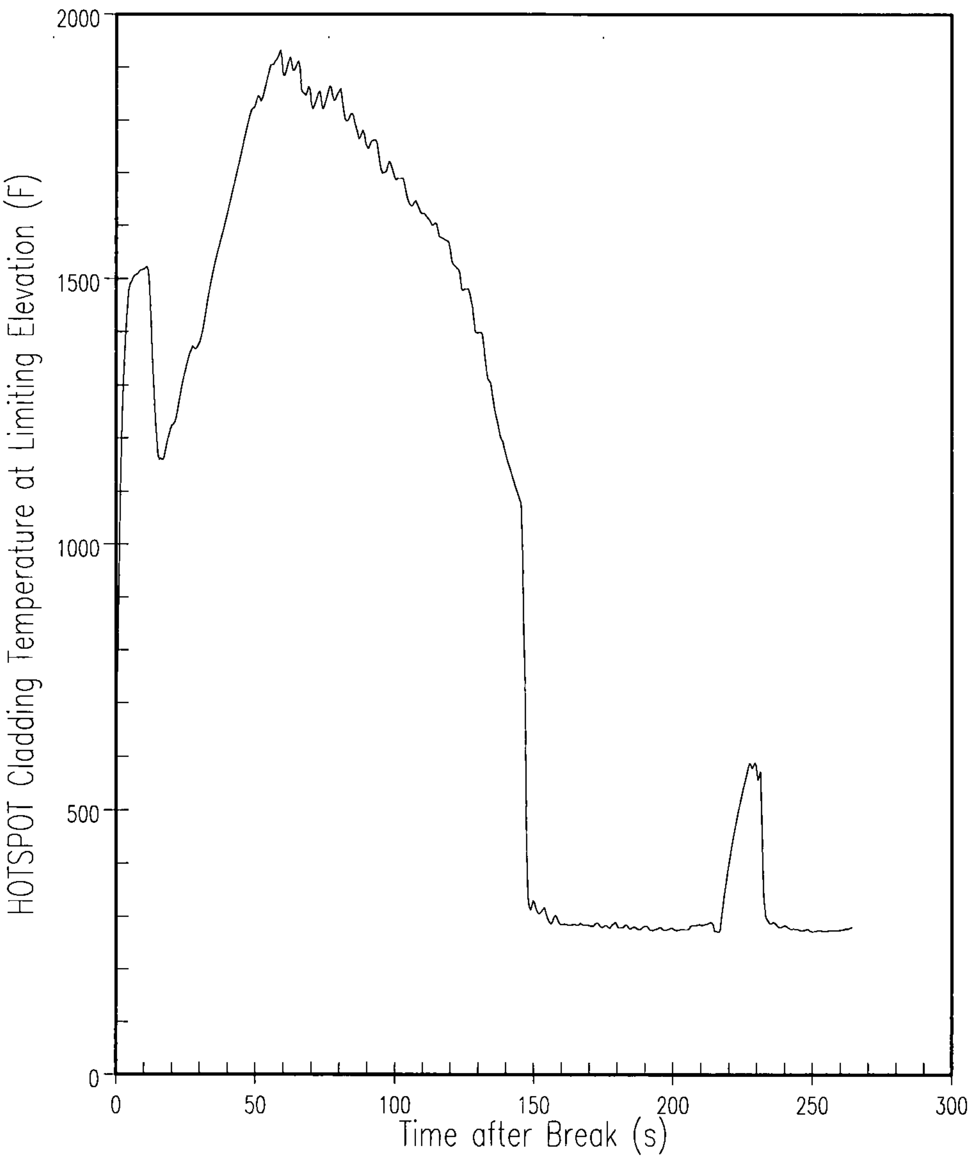

Figure 15.6.5.4A-2
HOTSPOT Cladding Temperature Transient at Limiting Elevation for 95th Percentile Estimator PCT Case

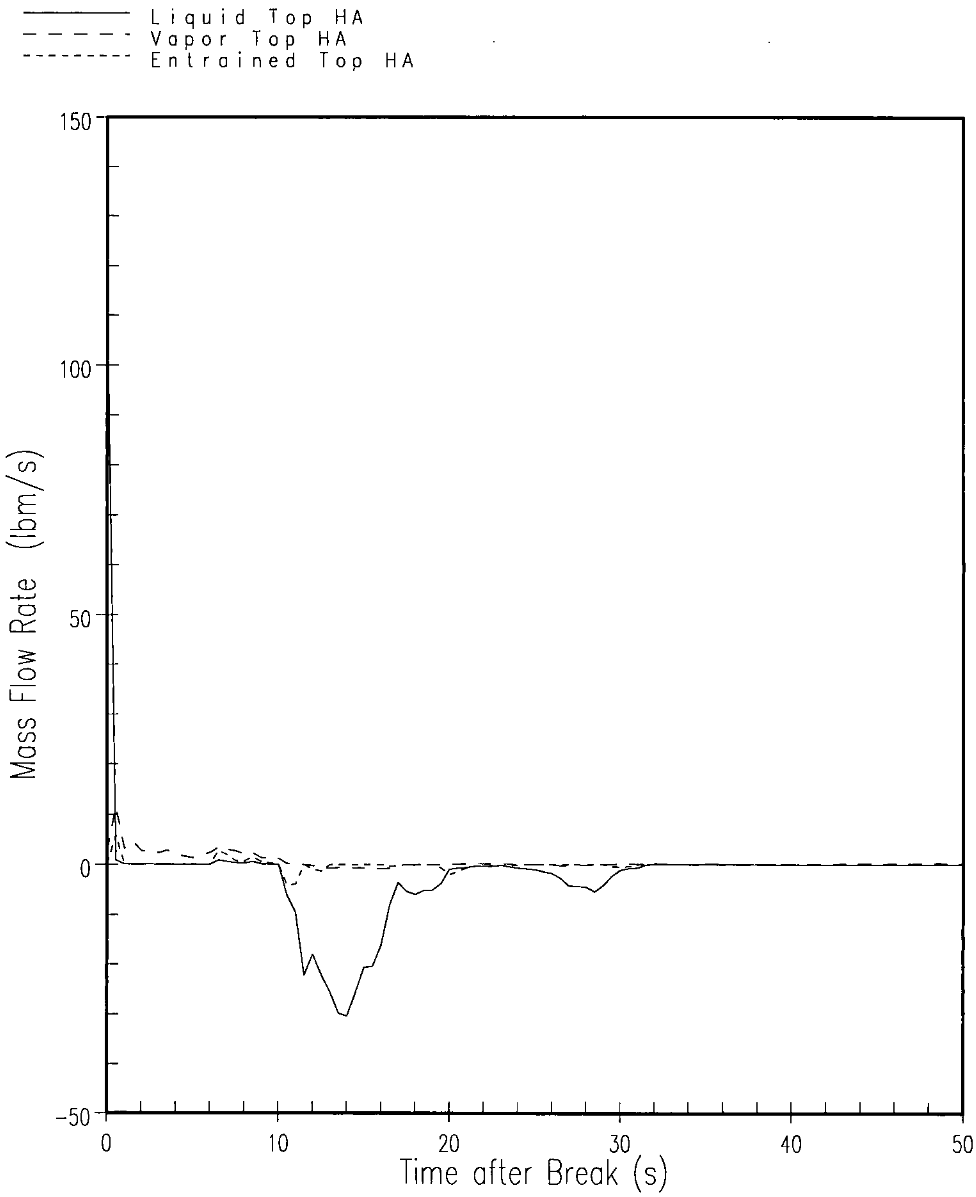

Figure 15.6.5.4A-3
Mass Flow at Top of Hot Assembly Channel for 95th Percentile Estimator PCT Case

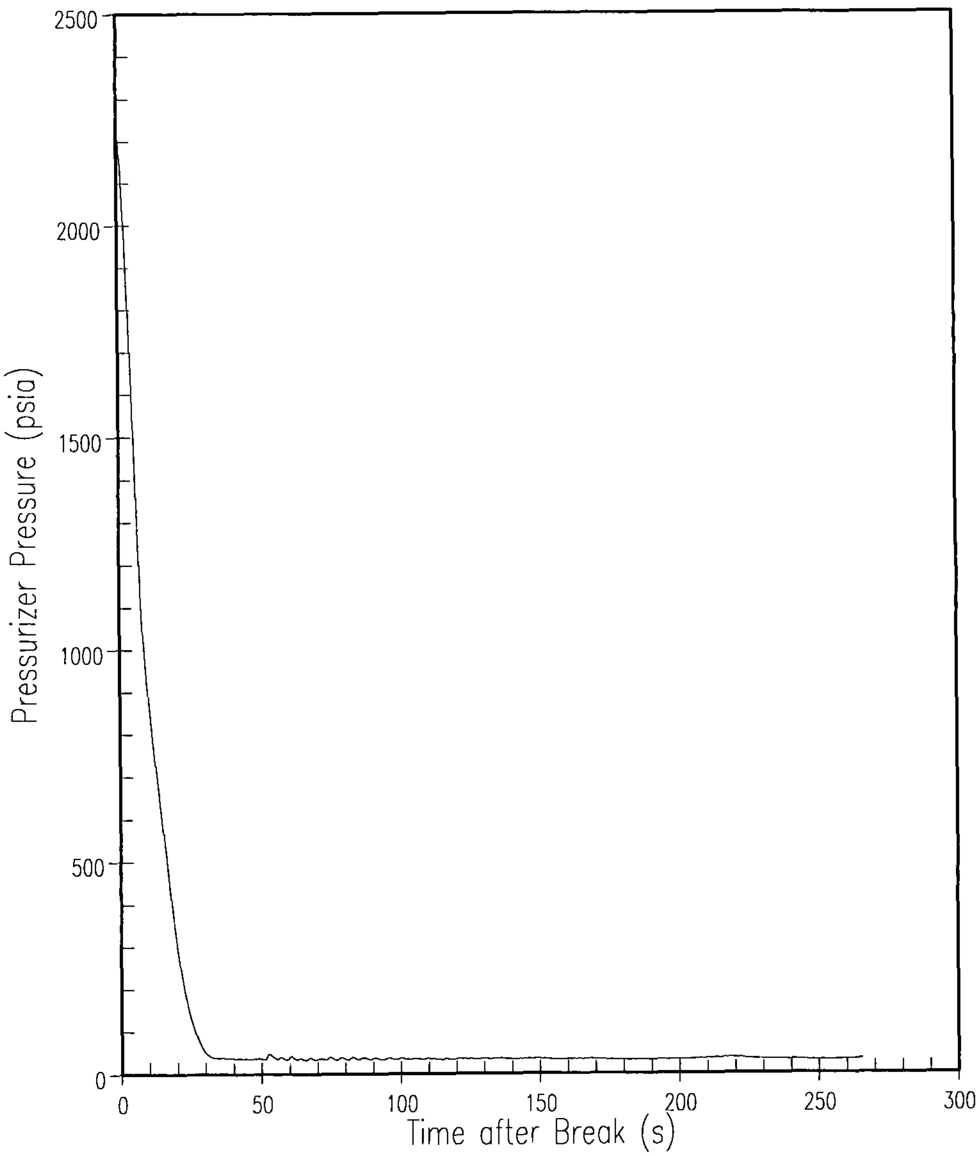

Figure 15.6.5.4A-4
Pressurizer Pressure for 95th Percentile Estimator PCT Case

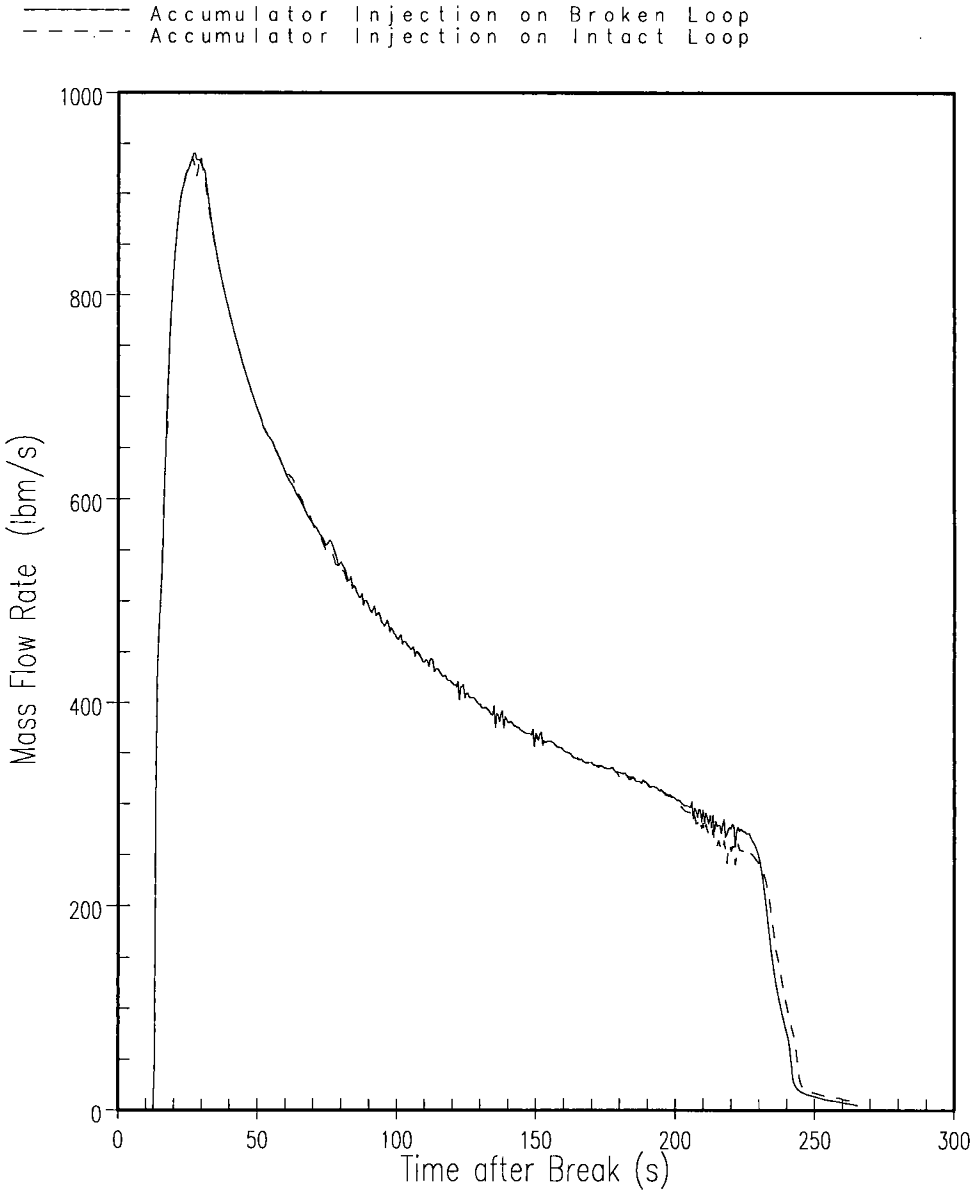

Figure 15.6.5.4A-5
Accumulator Injection Flow for 95th Percentile Estimator PCT Case

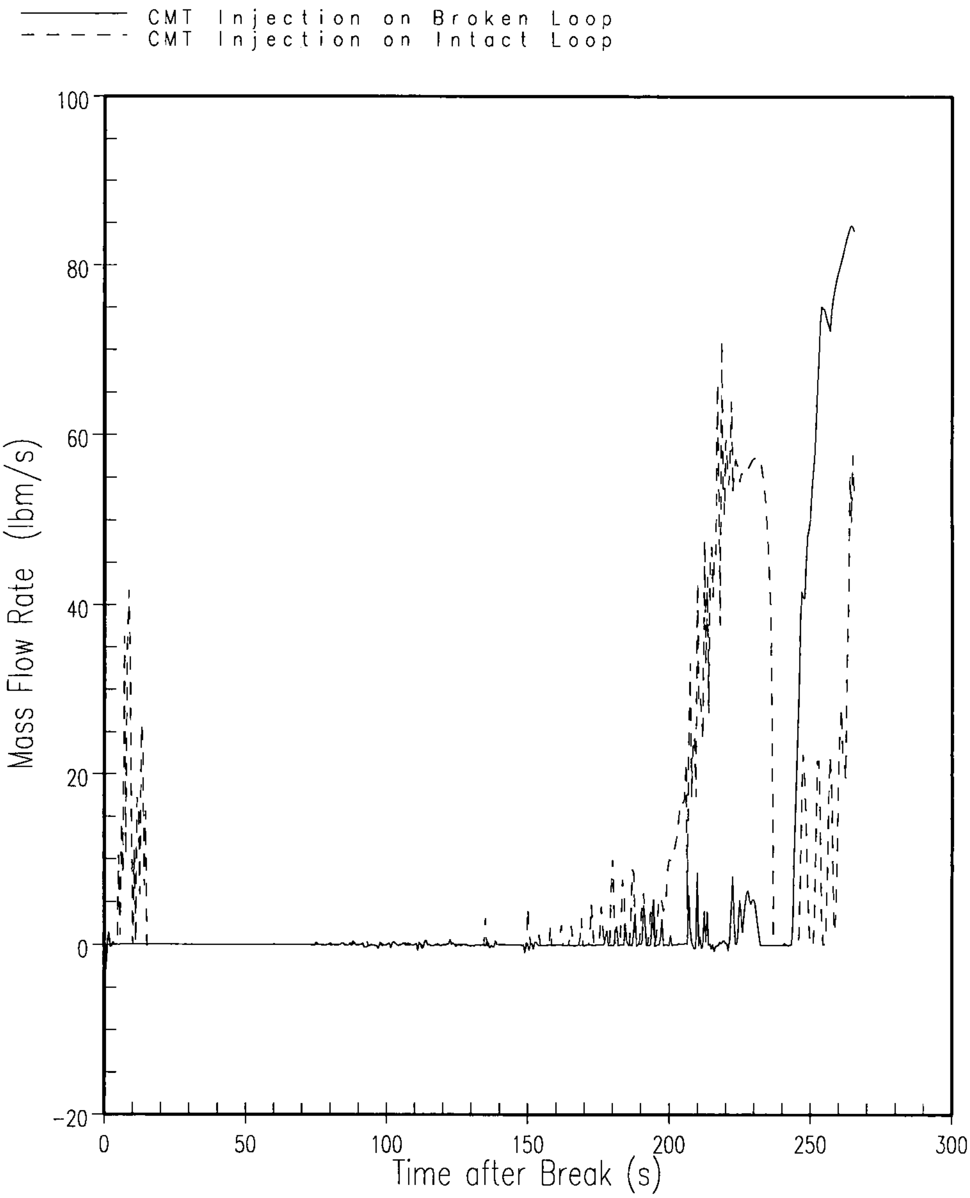

Figure 15.6.5.4A-6
Core Makeup Tank Injection Flow for 95th Percentile Estimator PCT Case

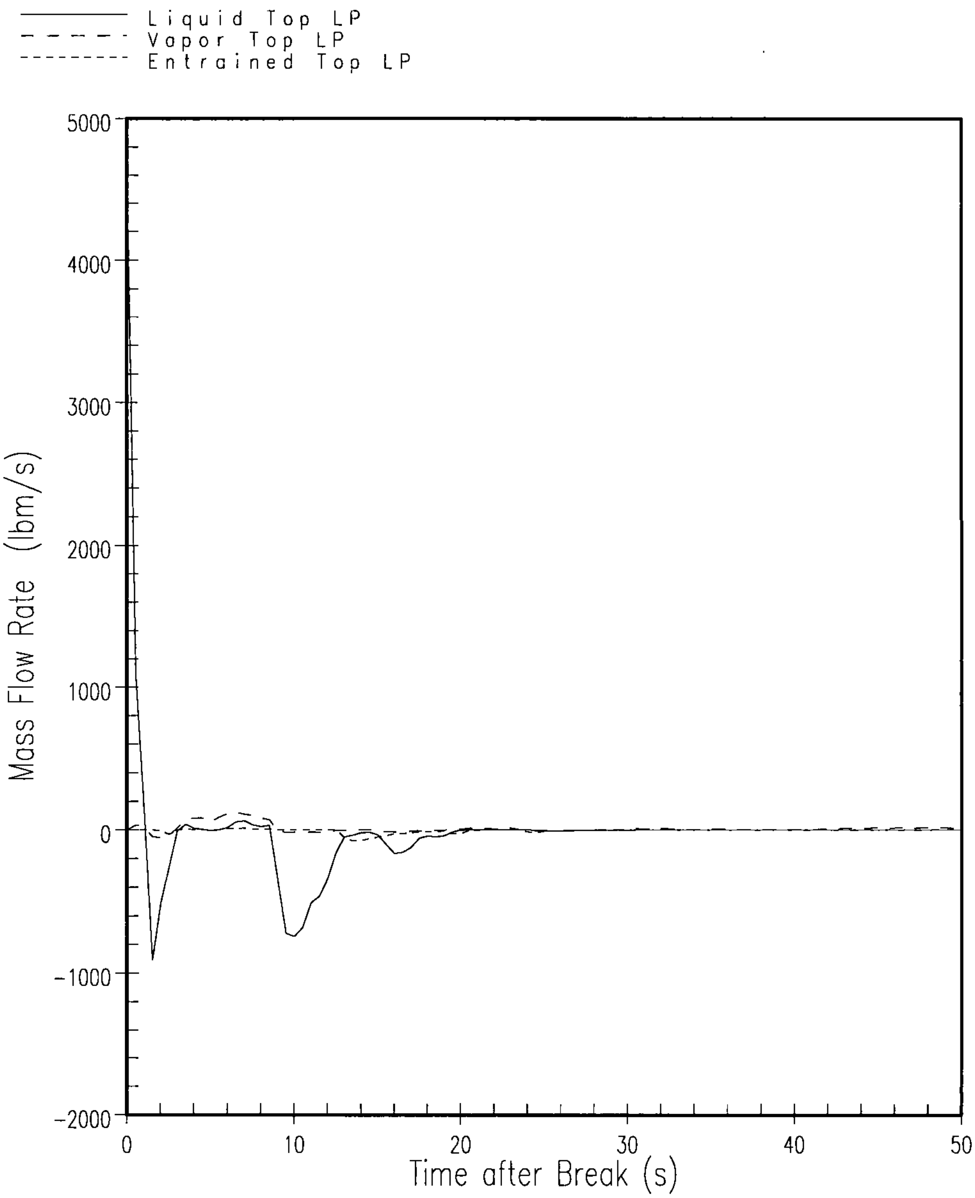

Figure 15.6.5.4A-7
Mass Flow at Top of Peripheral Assemblies Channel for 95th Percentile Estimator PCT Case

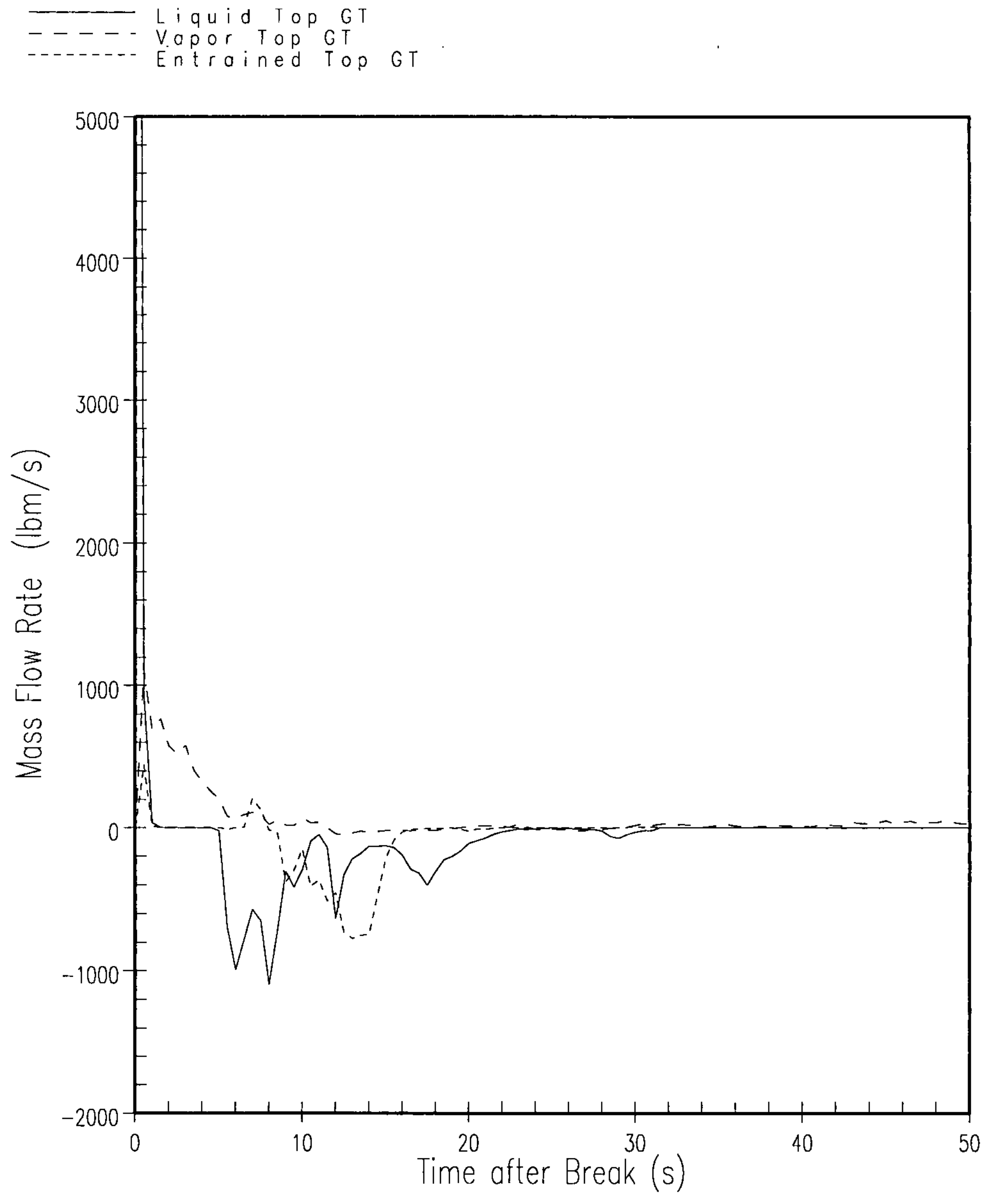

Figure 15.6.5.4A-8
Mass Flow at Top of Guide Tube Assemblies Chanel for 95th Percentile Estimator PCT Case

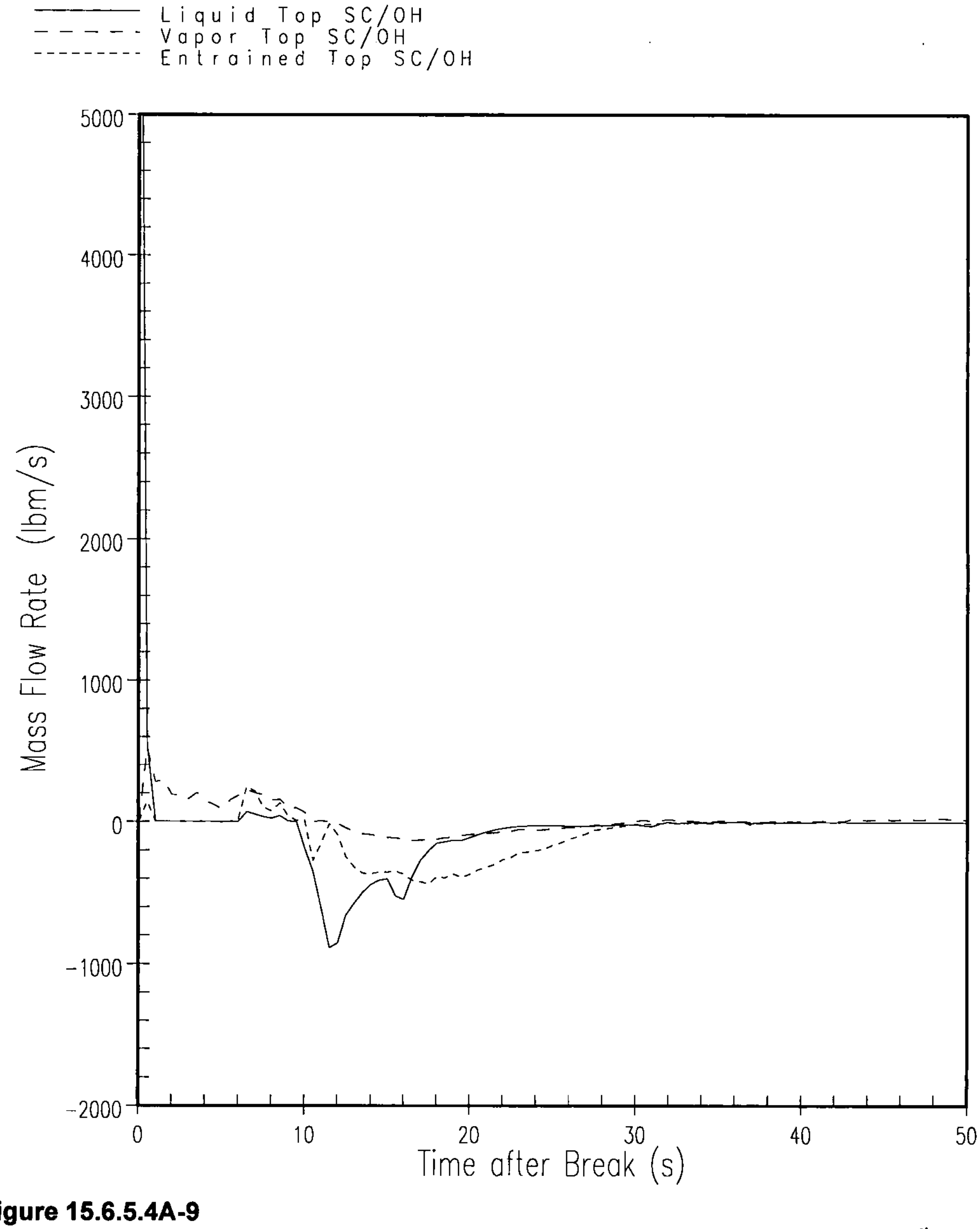

Figure 15.6.5.4A-9
Mass Flow at Top of Support Columns/Open Hole Assemblies Channel for 95th Percentile Estimator PCT Case

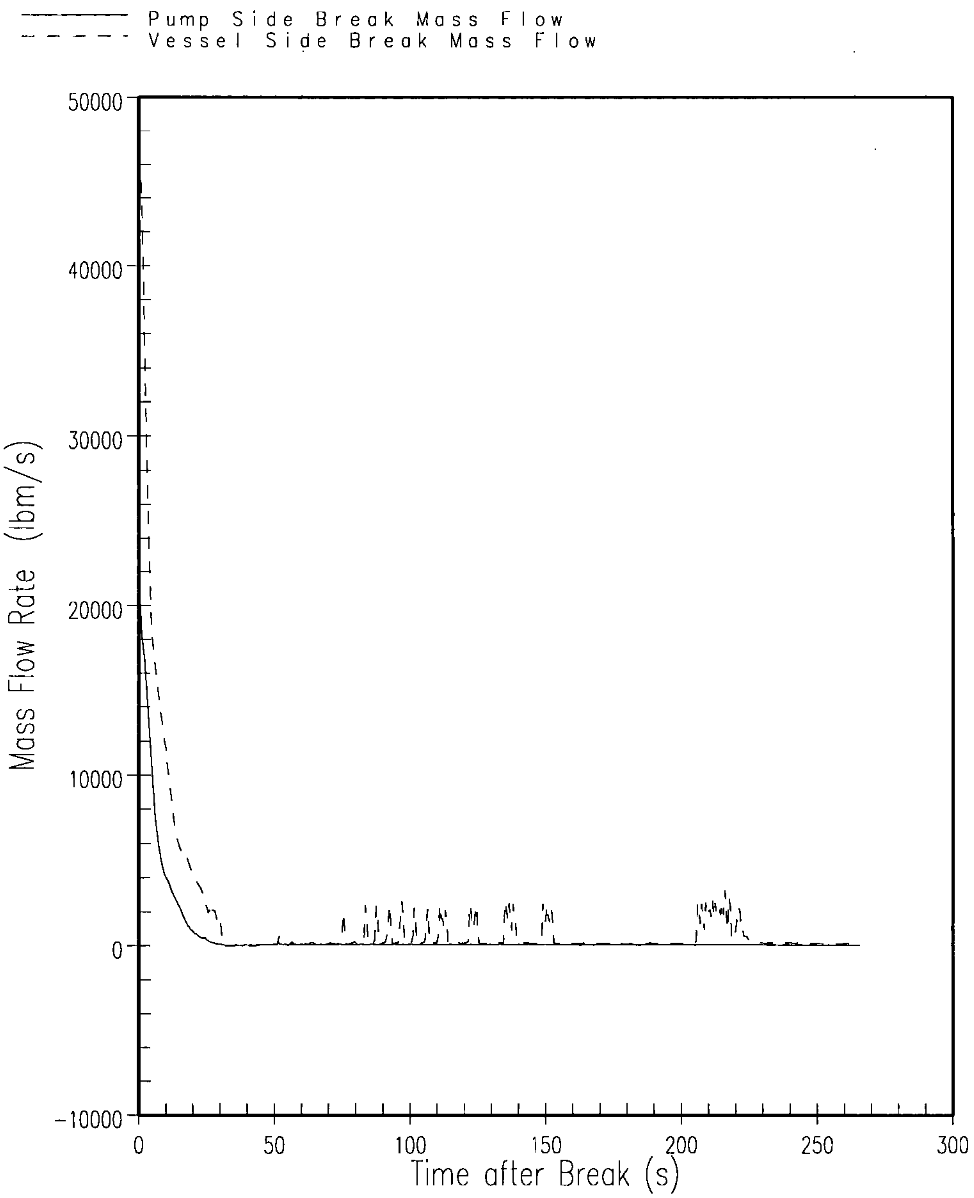

Figure 15.6.5.4A-10
Break Mass Flow for 95th Percentile Estimator PCT Case

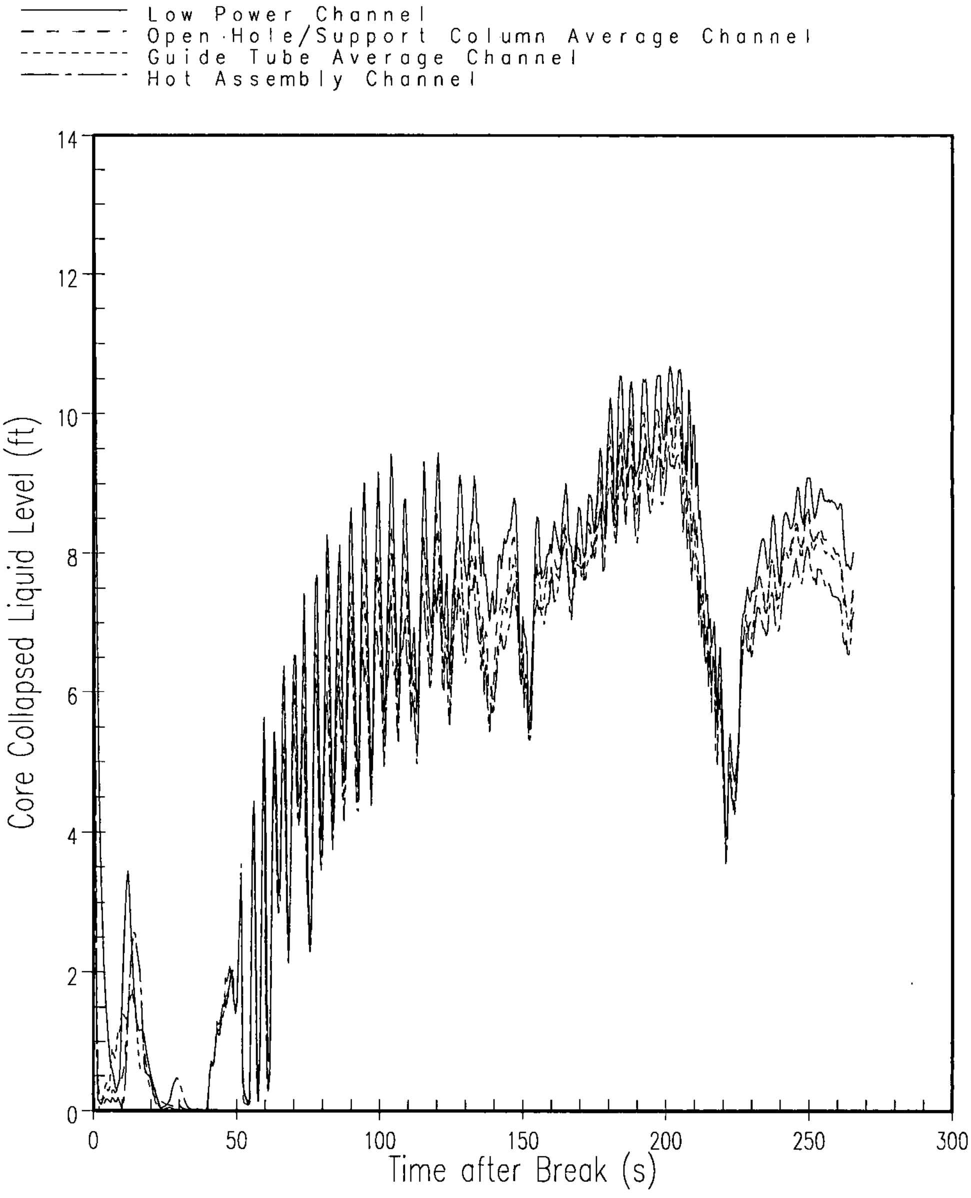

Figure 15.6.5.4A-11
Core Channel Collapsed Liquid Levels for 95th Percentile Estimator PCT Case (Reference Point: Bottom of Active Fuel)

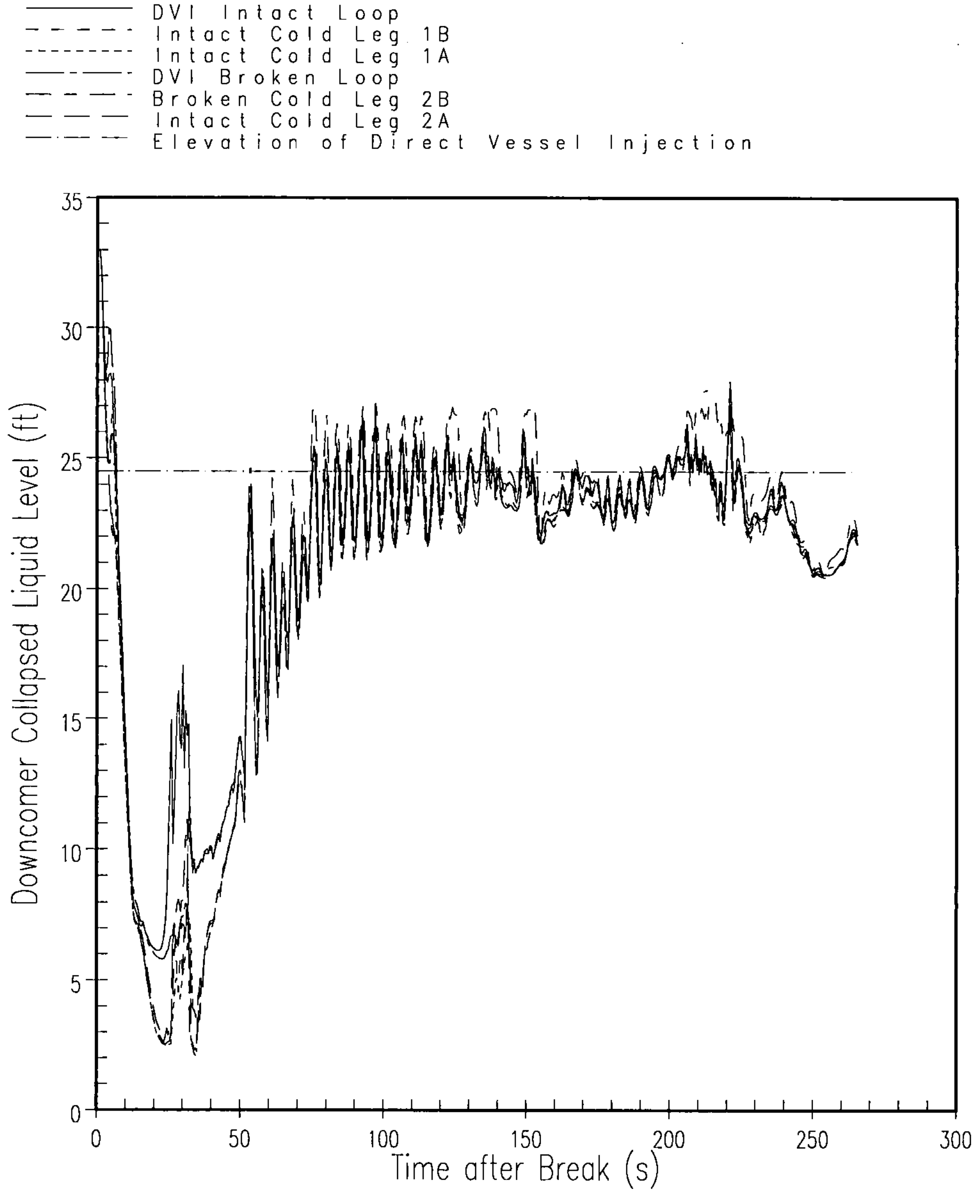

Figure 15.6.5.4A-12
Downcomer Channel Collapsed Liquid Levels for 95th Percentile Estimator PCT Case (Reference Point: Inside Bottom of Vessel)

CRR-009 *(LBLOCA Analysis)*

"It is stated in Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to address the Impact of Thermal Conductivity Degradation on Additional Events (LTR-NRC-12-56 P-Attachment), that the variations in rod internal pressure relative to burnup are already covered to a large extent in the small-break LOCA burnup studies with NOTRUMP."

"Please provide the small-break LOCA burnup studies referred in the Supplemental Information."

Westinghouse Response to CRR-009

The aforementioned statement from LTR-NRC-12-56 (Reference 1), "Variations in rod internal pressures relative to burnup are already covered to a large extent in the SBLOCA burnup studies…," is generic to all small break loss-of-coolant accident (SBLOCA) analyses performed using the NOTRUMP evaluation model (NOTRUMP-EM). Analyses that do not result in significant fuel rod heat-up, including the **AP1000** plant analysis, do not warrant burnup studies.

Fuel rod internal pressures increase throughout burnup and directly impact fuel rod burst prediction. Calculations performed at higher burnup steps may result in fuel rod burst and an associated cladding temperature spike. This is mainly due to the exothermic Zirc-water reaction on the fresh, inside surface of the cladding. The Zirc-water reaction is a function of temperature; as such, oxidation accruement and the associated heat release is negligible for plants with low peak cladding temperatures. The **AP1000** plant small break LOCA analysis (Reference 2) does not result in any fuel rod heatup; as such, fuel rod burst is not predicted to occur. Based on this, burnup studies for the **AP1000** plant SBLOCA analysis were not completed.

References

1. LTR-NRC-12-56, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to Address the Impact of Thermal Conductivity Degradation on Additional Events (Proprietary)," August 2012.

2. WCAP-17524-P, "AP1000 Core Reference Report," March 2012.

CRR-010 *(TH design)*

With respect to the AP1000 fuel design that was developed after the NRC approval of WCAP-15063-P-A, Revision 1, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0),"address the following:

a) Discuss the validation including the development of the thermal conductivity degradation (TCD) model used in the interim version of PAD 4.0, called PAD 4.0 TCD, including all coefficients of the TCD equation.

b) There are several versions of the TCD equation. What considerations were used for selecting the TCD equation used in the STAV 7.3 fuel performance code, which is licensed for BWR fuel but not for PWR, for the PAD 4 code?

c) Identify any modifications made to the STAV 7.3 TCD equation to make it compatible to the PAD 4.0 code and PWR fuel designs.

d) Include the plots of thermal conductivity versus temperature at burnups of 0, 20, 40, and 65 GWd/MTU with/without the burnup coefficients default option enabled.

e) Provide a discussion of the predicted PAD 4.0 data/Halden benchmark data set parameters including the initial/test conditions of the data and identify any deviations between the data sets conditions that may impact evaluation results between the data sets.

Westinghouse Response to CRR-010

a) Discuss the validation including the development of the thermal conductivity degradation (TCD) model used in the interim version of PAD 4.0, called PAD 4.0 TCD, including all coefficients of the TCD equation.

   The licensed PAD 4.0 fuel performance models (Reference 1) do not address the impact of fuel thermal conductivity degradation (TCD). However, TCD can be incorporated into the code using available input options. Fuel thermal conductivity is modeled in PAD using the following model form:

a,c

a,c

a,c

[ ]

The PAD 4.0 TCD assessment tool was then validated by comparison of PAD 4.0 TCD predicted centerline temperatures with measured fuel centerline temperatures for several Halden test reactor instrumented fuel assemblies (IFAs) with irradiation to high burnup. The PAD 4.0 TCD validation included data from the following Halden test programs: [

]^{a,c} The PAD 4.0 TCD predicted fuel centerline temperatures were found to be in good agreement with measured values with no further adjustment to the TCD model coefficients.

Additional details of the PAD 4.0 TCD fuel centerline temperature validation are provided in the response to CRR-010, Part (b).

b) In the development of the PAD 4.0 TCD assessment tool, the STAV TCD model based on the Halden TCD model form and the modified NFI model form currently incorporated into the FRAPCON 3.4 NRC fuel performance audit code were considered. A comparison of these models, as documented in the Technical Evaluation Report generated by Battelle Pacific Northwest National Laboratory (PNNL) for the NRC review of the STAV 7.2 code (Reference 4), confirmed that there is excellent agreement between these two independent models. Given this good agreement, [

]^{a,c}

[

]^{a,c} Furthermore, the acceptability of the model form and associated coefficients was confirmed by comparison of predicted and measured fuel centerline temperatures.

c) The fuel thermal conductivity equation in the STAV code is of the following form:

a,c

[ ]

The only other modifications were to convert the constants to account for the differences in units for K_{UO2} and burnup. For K_{UO2}, the STAV units are W/m·°C while the PAD 4.0 units are W/cm·°C, and for burnup the STAV units for u are MWd/kgUO_2 while the PAD 4.0 units for Bu are GWd/MtU. No other changes were required to make the STAV f(Bu) TCD function compatible with the PAD 4.0 f(Bu) function in the thermal conductivity equation.

d) Plots of fuel thermal conductivity versus temperature, with and without TCD, using the PAD 4.0 conductivity model are provided below for 0.0, 20.0, 40.0 and 65.0 GWd/MtU in Figures 1 through 4, respectively.

a,c

Figure 1: PAD 4.0 and PAD 4.0 TCD Fuel Thermal Conductivity versus Temperature at 0.0 GWd/MtU

a,c

Figure 2: PAD 4.0 and PAD 4.0 TCD Fuel Thermal Conductivity versus Temperature at 20.0 GWd/MtU

a,c

Figure 3: PAD 4.0 and PAD 4.0 TCD Fuel Thermal Conductivity versus Temperature at 40.0 GWd/MtU

a,c

Figure 4: PAD 4.0 and PAD 4.0 TCD Fuel Thermal Conductivity versus Temperature at 65.0 GWd/MtU

e) The PAD 4.0 fuel performance code thermal model was calibrated and validated based on data from three NRC sponsored Halden tests: IFA-431, IFA-432 and IFA-513. Each of these tests had six test rodlets fabricated at the Battelle Pacific Northwest National Laboratory (PNNL). The test rod fabrication parameters were designed to test a range of initial pellet-clad gap conditions (from 2.2 mils to 15.1 mils) that bound the initial pellet-clad gaps used in commercial fuel rod design. The IFA-431 and IFA-432 test rods were pre-pressurized with one atmosphere helium. The IFA-513 test rods had a range of pre-pressurization conditions that included 100% helium at one or three atmospheres and rods with one atmosphere pre-pressurization with a mixture of helium and argon. All of the test rods were instrumented with thermocouples at the top and bottom of the fuel stack.

On-line fuel centerline temperature measurements obtained as a function of burnup for rod burnup up to 5 GWd/MtU were used in PAD 4.0 and prior PAD 3.4 thermal model development. The range of initial gap conditions spanned the range of gap conditions expected in Westinghouse PWR operation and these data were therefore considered applicable to the whole operating burnup range. By focusing on the relatively low burnup data, uncertainties associated with cladding creep, fission gas release and thermocouple decalibration effects were minimized.

The IFA-432 test rig was irradiated to extended burnup in excess of 45 GWd/MtU, and the Halden project developed decalibration correction factors to account for the expected irradiation effects on the thermocouple response. The PAD 4.0 thermal model was further validated using the higher burnup data for the IFA-432 test rods. The measured versus predicted fuel temperature comparisons for the extended burnup data did not show a clear trend with burnup and the PAD 4.0 thermal model was accepted by the US NRC without fuel thermal conductivity degradation.

Subsequent to these NRC sponsored Halden tests, additional experiments have been conducted to further understand fuel properties as a function of burnup. To avoid uncertainties associated with thermocouple decalibration, two alternative approaches were considered. First, tests were run with annular pellet fueled rods with expansion wire thermometers in the annulus. This instrumentation is not subject to decalibration under irradiation, and these tests provided the first definitive results confirming TCD. The expansion wire thermometer results could, however, be subject to error due to the potential for pellet fragment relocation into the annulus and possible interference with the wire. Additional tests were therefore performed using fuel rods that were commercially irradiated to high burnup and then re-fabricated into test rods instrumented with fresh thermocouples. These rods provide high quality temperature measurements at high burnup, but with some additional uncertainty in the fuel rod conditions due to re-fabrication. All of these types of testing were used to validate the PAD 4.0 TCD model.

Data from the following tests were added to the PAD thermal model database:

- [

]^{a,c}

[

]a,c

The database used to validate the PAD 4.0 TCD assessment tool for fuel temperature predictions with TCD therefore included test data obtained from [

]a,c at high burnup. The combination of all of these data provide a substantial database that bounds both open gap and closed gap operation conditions.

References

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

2. W. Wiesenack and T. Tverberg, "Thermal Performance of High Burnup Fuel – In-Pile Temperature Data and Analysis," International Topical Meeting on Light Water Reactor Fuel Performance, April 10-13, 2000.

3. WCAP-15836-P-A, Revision 0, Volume 1, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," April 2006.

4. WCAP-15836-P-A, Revision 1, Volume 1, Section B, sub-section 2.1, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," April 2006.

CRR-011 *(TH design)*

In the PAD 4.0 code, the thermal conductivity correlation is a function of [ ]^{a,c} The burnup dependent thermal conductivity degradation equation from STAV 7.3 is used in the interim version of PAD 4.0, called PAD 4.0 TCD. The STAV 7.3 code is approved for BWR fuel design analysis and is designed to handle fuel with gadolinium as an integral burnable absorber. The **AP1000** PWR fuel design may include axial blankets (fuel pellets of a reduced enrichment), annular fuel pellets in the top and bottom 8 inches of the fuel stack (fully enriched or partially enriched), and integral fuel burnable absorbers (boride-coated fuel pellets or fuel pellets containing gadolinium oxide mixed with uranium oxide). Address the following:

a) What are the impacts that these fuel design features may have on the thermal conductivity correlation?

b) Provide plots of the PAD 4 TCD predicted temperature profile data and the Halden benchmark data with/without the fuel burnup coefficient enabled.

c) Discuss any differences between the data profiles.

d) Discuss the applicability of the burnup dependent thermal conductivity degradation equation from STAV 7.3 for IFBA coated pellets.

Westinghouse Response to CRR-011

a) The Halden test rods used for thermal model validation are applicable to the full range of fuel pellet types used in the **AP1000** PWR fuel rod design. Annular pellets are included in the Halden tests, and the test results are therefore applicable to annular axial blanket fuel pellets. The UO_2 material properties are the same for the full range of U-235 enrichments proposed for use in the **AP1000** PWR fuel rods, including the use of natural enriched uranium in the axial blanket pellets. The effects of enrichment on pellet radial power distributions are accounted for as input to the PAD code. The PAD 4.0 TCD model validation is therefore applicable to the range of pellet types used in the axial blanket region. The ZrB_2 Integral Fuel Burnable Absorber (IFBA) pellets have a [ ]^{a,c} coating of ZrB_2 on the outside pellet surface. This coating has only a negligible thermal resistance and is conservatively modeled in PAD as a small increase in the fuel diameter. The coating has no impact on the temperature gradient across the fuel pellet, and the Halden test results for uncoated pellets are applicable to the ZrB2 IFBA design. Gadolinia bearing IFBA fuel pellets are also an option in the **AP1000** PWR fuel rod design. The inclusion of gadolinia in the fuel matrix acts as an impurity and the unirradiated thermal conductivity of the Gd2O3-UO2 fuel is appropriately reduced as a function of the gadolinia content. [ ]^{a,c} in addition to the reduction in the unirradiated fuel thermal conductivity due to the gadolinia. The fission product generation in Gd_2O_3-UO_2 fuel is similar to that for UO_2 fuel and no further changes are required to appropriately

assess TCD in gadolinia fuel. In summary, the PAD 4.0 TCD assessment tool is applicable for use with all fuel types considered in the **AP1000** PWR core design.

b) Provided below are plots of measured minus predicted fuel centerline temperature as a function of rod average burnup for [ ]a,c based on PAD 4.0 without TCD and based on the PAD 4.0 TCD assessment tool. These results are typical for the high burnup test rods.

a,c

[ ]

a,c

[ ]

c) Figures 1 and 2 are shown with the same scale on the Y-axis to more clearly show the differences between the PAD 4.0 and PAD 4.0 TCD comparisons. The PAD 4.0 comparison in Figure 1 indicates a steadily increasing difference in the measured minus predicted temperature as a function of burnup as would be expected due to TCD. When the STAV TCD model coefficients are input to PAD 4.0 TCD, the measured minus predicted difference becomes relatively flat as a function of burnup, as shown in Figure 2. The only difference between the two prediction sets is the use of the PAD 4.0 TCD model [

]^{a,c} by PAD 4.0.

d) In the Westinghouse ZrB_2 IFBA coated pellet design, a coating of ZrB_2 material is applied on the outside cylindrical surface of the pellet. The coating thickness is [

]^{a,c}. In the PAD thermal model, the coating is conservatively modeled based on the assumption that the coating [ ]^{a,c}. However, the coating is distinct from the fuel pellet and does not change the properties of the base UO_2 pellet material. The coating does not impact the pellet radial power distribution and therefore the burnup distribution and the associated radial fission product distribution through the pellet is not impacted by the coating. The presence of a thin ZrB_2 coating does not change the thermal conductivity within the fuel pellet. Therefore, the TCD impact established based on non-IFBA pellet fuel temperature measurements is applicable to the IFBA pellet.

CRR-013 *(GSI-191)*

WCAP-17524-P includes a description of fuel design changes to be included in the advanced first core for the AP1000 plant design. Section 2.6 discusses the impacts of the protective grid design change on GSI-191 but does not discuss the impacts of other fuel design changes. Provide an analysis of the impacts on GSI-191 for each design change, including the eIFM and eMid-Grids.

Westinghouse Response to CRR-013

The justification for limiting the evaluation of the GSI-191 issue to the protective grid is based on the statement in Section 2.6 of WCAP-17524, *AP1000 Core Reference Report* (Reference 1); *IMPACTS OF PROTECTIVE GRID DESIGN CHANGE ON GSI-191*, [

]^{a,c} As debris is captured in the fuel assembly, the overall fuel assembly ΔP increases, and testing has demonstrated that [

]^{a,c} is a result of the relatively low core flow rates associated with the **AP1000** PWR Cold Leg breaks which are effectively driven by the natural circulation conditions. The collection of the debris [

]^{a,c}

As a result of the lengthy process that has evolved as the industry and the NRC have worked to close out the GSI-191 issue, Westinghouse put in place an "interim" method for evaluating changes to the fuel components. This process helps ensure that new fuel features can be properly evaluated, and tested if necessary, for the GSI-191 issue. [

]^{a,c}

As noted in the **AP1000** PWR Core Reference Report (WCAP-17524), the final **AP1000** PWR IFM and Mid-grids have [ ]^{a,c} when compared to the IFM and Mid-grids of the **AP1000** PWR fuel assembly described in the **AP1000** PWR Design Control Document. This is also shown in Columns 2 and 3 of Tables 1 and 2. The IFM has about [ ]^{a,c}

[ ] whereas the Mid-grid has about [ ]^{a,c}, relative to that presented in the **AP1000** PWR Design Control Document for the IFMs and Mid-grids. This [

]^{a,c}, as shown in Columns 2 and 3 of Tables 1 and 2, is [
]^{a,c}. Features such as the grid thickness for the inner and outer straps, remain unchanged. Column 4 of Tables 1 and 2 shows the [
]^{a,c} (Reference 3) The fifth column shows the
[
]^{a,c} (see Reference 3). These differences are acceptable when an examination of the GSI-191 performed for the Robust Protective Grid (RPG) is performed, as explained below.

To demonstrate the acceptability of the RPG, GSI-191 tests were specifically run for a "representative" **AP1000** PWR Cold Leg test case which had been previously run with a Standard Protective Grid. The intent was to demonstrate that under the **AP1000** PWR post-LOCA conditions that a fuel assembly with the RPG would be no more limiting than a fuel assembly with the Standard Protective Grid. Figure 1 shows the results from one of these "representative" **AP1000** PWR Cold Leg test cases with a fuel assembly with the RPG. The [
]^{a,c} than the GSI-191 test results for the fuel assembly with the Standard Protective Grid, which had a [ ]^{a,c}

A close examination of Figure 1 shows that [ ]^{a,c} of the test fuel assembly, which is shown by the green line. The magnitude of the pressure drop across the [ ]^{a,c} which is essentially [ ]^{a,c} when compared to the total fuel assembly ΔP of [ ]^{a,c} (shown by the aqua line). [
]^{a,c} which is shown by the red line on Figure 1. [

]^{a,c} for this GSI-191 "representative" case, [
]^{a,c} (which is based on the GSI-191 test results when the standard Protective grid was used).

Given the above, the final **AP1000** PWR IFM and Mid-grids presented in WCAP-17524-P would not result in a condition that would adversely affect the GSI-191 issue as it applies to the **AP1000** plant. It can therefore be concluded that long term core cooling would be satisfied for the **AP1000** PWR fuel assembly described in WCAP-17524-P.

Table 1: Comparison of Key Mid-Grid Characteristics

a,c

Table 2: Comparison of Key IFM Grid Characteristics

a,c

a,c

Figure 1: Fuel Assembly ΔP from Representative **AP1000** PWR GSI-191 Test

References

1. WCAP-17524-P, *AP1000 Core Reference Report*, March 1, 2012.

2. APP-GW-GL-700, Revision 19, *AP1000 Design Control Document*, June 13, 2011.

3. WCAP-17028-P, Revision 6, *Evaluation of Debris-Loading Head-Loss Tests for AP1000™ Fuel Assemblies During Loss of Coolant Accidents*, June 2010.

CRR-014 *(GSI-191)*

In order to support the staff's review of the new fuel assembly design impacts on GSI-191, provide a comparison of the fuel designs for the following:

a) AP1000 DCD Rev. 19

b) Test assembly used to support AP1000 DCD Rev. 19

c) Fuel assembly design for WCAP-17524-P

d) Test assembly for PWROG which included the Robust P-Grid.

Also provide the test conditions for both the AP1000 DCD Rev. 19 tests and the PWROG tests used to support the Robust P-Grid (e.g. flow rates, debris types/quantities, etc.).

Westinghouse Response to CRR-014

Table 1 presents a detailed comparison of the different key parameters/dimensions for the individual components of the fuel designs and the test conditions for which the GSI-191 testing was performed.

Regarding the test conditions for the **AP1000** DCD Revision 19 fuel assembly, Section 7 *Test Matrix and Initial Conditions* of WCAP-17028-P, Revision 6 (Reference 1) presents detailed information covering the various conditions using in the testing. A total of 39 different cases are presented.

To test the Robust Protective Grid for the **AP1000** plant, the test conditions presented in Table 14-2 were assumed. These are the test conditions described in Section 2.6 *Impacts of Protective Grid Design Change on GSI-191* of WCAP-17524-P (Reference 2).

Table 1: Comparison of Key Parameters for Different Fuel Designs Associated with AP1000 and GSI-191 Testing

a,c

Table 2: Summary of GSI-191 Test Conditions for the AP1000 Plant Cold Leg Break Type Cases with the Robust Protective Grid

a,c

References

1. WCAP-17028-P, Revision 6, *Evaluation of Debris-Loading Head-Loss Tests for AP1000™ Fuel Assemblies During Loss of Coolant Accidents*, June 2010.

2. WCAP-17524-P, *AP1000 Core Reference Report*, March 1, 2012.

CRR-015 *(GSI-191)*

What is the screening methodology used by Westinghouse to determine the impacts on GSI-191 for each fuel design change.

Westinghouse Response to CRR-015

The screening method that Westinghouse has used (and is the current screening methodology) to determine the impacts of GSI-191 on the various fuel components is based on which fuel component is being modified as different components have a higher propensity for capturing/trapping debris. For example, for fuel components such as [ ]a,c the fuel assembly. On the other hand, fuel components such as the [ ]a,c are more easily dispositioned as "changes" as these components do not in general affect the capture of debris on the fuel assembly.

To ensure that all newly proposed fuel assembly components are properly evaluated, Westinghouse has developed a screening method for the GSI-191 issue. In this screen method, most any change to the [ ]a,c For the grids, the screening methodology entails an examination of the [ ]a,c A final assessment is then made based on a summary of the ranked changes which would then possibly lead to specific GSI-191 testing. This screening method is dependent on the specific fuel/plants being evaluated. For example, as noted in the response to CRR-013, the **AP1000** PWR GSI-191 testing demonstrated that [ ]a,c

In conclusion, fuel component design changes that are proposed/considered must be evaluated for their potential impact on the GSI-191 issue. To ensure that fuel component changes are properly evaluated, Westinghouse has defined a screening process that examines changes to key component characteristics, as well as plant specific information [ ]a,c to determine if specific GSI-191 testing is needed to support the change(s).

CRR-017 *(thermal conductivity degradation)*

The PAD 4.0 code incorrectly excludes burn-up effects in its thermal conductivity model. By using a corrected thermal conductivity model and the existing fission gas model, the code could overestimate the amount of fission gas and the fuel rod internal pressure since the current fission gas model was correlated to the testing data using the original thermal conductivity model. To correct this, [

]^{a,c} it is unknown if the calculated amount of fission gas released remain conservative for **AP1000** PWR applications, considering the differences in geometry, burn-up levels, and power history between the test assemblies used to calculate the fission gas release model and the **AP1000** Core Reference Report application design. Demonstrate that this assumption will lead to conservative or accurate predictions of fission gas quantities and fuel rod internal pressure within the power and burn-up range of this application.

Westinghouse Response to CRR-017

This question addresses whether the PAD 4.0 (Reference 1) fission gas release model database reasonably bounds the specific fuel conditions of the **AP1000** PWR fuel rod design in the **AP1000** Core Reference Report. The fission gas release model database for PAD 4.0 includes measured fission gas release data for more than 250 fuel rods. The database includes rods that operated under steady state conditions throughout their life and rods that were subjected to transient power increases. The steady state fission gas release rods includes rods that were irradiated under typical commercial reactor operating conditions as well as test rods that operated at steady state powers well in excess of commercial reactor conditions.

The range of fabrication and operating parameters covered by the steady state rods in the fission gas release database is summarized in the following Table 1 along with expected **AP1000** PWR fuel rod parameters based on the Core Reference Report. The **AP1000** PWR fuel rod design and projected operating duty is very similar to that which is experienced by current operating Westinghouse 17X17 (0.374" OD) fuel, and the **AP1000** PWR design and operating parameters are bounded by the PAD 4.0 fission gas release model database parameter ranges.

The fuel rod average burnups in the PAD 4.0 steady state fission gas release database range up to ~69 GWD/MTU. Figure 1 illustrates the PAD 4.0 steady state measured fission gas release as a function of rod average burnup. This figure clearly illustrates that there is a substantial number of fission gas release measurements at elevated burnups. [

]^{a,c}

The PAD 4.0 fission gas release model has been developed based on a database that bounds all of the key parameters important for fission gas release for the **AP1000** PWR fuel rod design, and [

includes the impact of TCD. This NRC approved model is therefore an appropriate and conservative basis for prediction of fission gas release for the **AP1000** PWR design when evaluating the impacts of TCD.]^{a,c}

Table 1: Comparison of **AP1000** Fuel Fabrication and Operation Parameters with PAD 4.0 Fission Gas Release database Parameter Ranges

Parameter	Range in PAD 4.0 Fission Gas Release Model	**AP1000** CRR Parameter
Fuel Density (% T.D.)	[ ]^{a,c}	95.5
Initial Pressure, psia (cold)	[ ]^{a,c}	100 – 275
Initial Gas Composition	[ ]^{a,c}	Helium plus one atmosphere air
Fuel pellet diameter, inches	[ ]^{a,c}	0.3225
Cold Initial Pellet-Clad Gap, inches	[ ]^{a,c}	0.0065
Cladding Outer Diameter, inches	[ ]^{a,c}	0.374
Rod Average Burnup, MWD/MTU	[ ]^{a,c}	≤ 62000
Rod Time Average Linear Power, kW/ft	[ ]^{a,c}	~7.5
Peak Rod Average Linear Power, kW/ft	[ ]^{a,c}	9.8
Peak Local Linear Power, kW/ft	[ ]^{a,c}	14.8

a,c

Figure 1 PAD 4.0 Measured Fission Gas Release as a Function of Rod Average Burnup

References

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

CRR-019 *(non-LOCA accidents)*

The rod ejection accident was analyzed using the previous fuel pellet thermal conductivity model, which does not account for burnup effects. With the new pellet thermal conductivity model, it is expected that the fuel pellet initial steady state temperature is significantly higher than what was calculated before, especially for the end-of-cycle condition. What is the quantitative impact of the corrected TCD model with burnup dependence on the rod ejection accident analysis? In particular, provide the impact on the most limiting DNBR, fuel center line temperature and cladding strain. Demonstrate that they all satisfy the relevant limits.

Westinghouse Response to CRR-019

For the Core Reference Report (CRR), the rod ejection analysis of each criterion is divided into two distinct parts: the modeling of the nuclear power transient and the modeling of the hot rod transient response. The modeling of the nuclear power transient and the amount of fission energy deposited into the fuel already includes the effects of TCD in the calculation of the fuel temperatures used for Doppler feedback effects. The hot rod transient response is further divided into two distinct models: one calibrated to produce maximum heat flux / minimum DNBR and the other to produce minimum heat flux / maximum fuel temperature and enthalpy.

The hot rod model to maximize heat flux (minimize DNBR) has been calibrated to [ ]^{a,c} used throughout all the safety analyses. For a given amount of fission energy deposited in the fuel, the heat flux out of the fuel rod is maximized by [ ]^{a,c}. This is accomplished by calibrating the DNB hot rod model using [ ]^{a,c}. Decreasing the fuel thermal conductivity (e.g. by including TCD effects) would decrease the heat transfer through the pellet thereby reducing the heat flux from the fuel rod and increasing the DNBR.

This point is illustrated by Figure 1, which shows the minimum DNBR and core nuclear power versus time. The DNBR is plotted for two different hot rod models assuming identical nuclear power transient inputs. The standard DNBR hot rod model is calibrated to the [ ]^{a,c} as described above. The "best estimate" DNBR model is calibrated to the [ ]^{a,c}. While both models start the transient with the same DNBR, the "best estimate" model results in a higher DNBR due to [ ]^{a,c}. This demonstrates that excluding TCD effects from the hot rod DNBR analysis yields conservative results, and that the results presented in the CRR remain valid.

The hot rod model to minimize heat flux (maximize fuel temperature and enthalpy) has been calibrated to [ ]^{a,c} used throughout all the safety analyses. For a given amount of fission energy deposited in the fuel, the heat flux out of the fuel rod is minimized by [ ]^{a,c}

[ ]^{a,c}. Decreasing the fuel thermal conductivity (e.g. by including TCD effects) would further decrease the heat transfer through the pellet thereby reducing the heat removed from the fuel rod and increasing the fuel temperature.

The primary effect of TCD on the CRR rod ejection peak fuel temperatures and enthalpies will be in the initial fuel temperature / enthalpy conditions of the fuel rod. This is because the enthalpy hot rod model used for the CRR analysis did not initially include TCD effects. However, as previously stated, the nuclear power transient and the amount of fission energy deposited into the fuel already included the effects of TCD, therefore the impact on the [ ]^{a,c}. This is illustrated by the enthalpy and fuel temperature results with and without TCD effects presented in Table 1. As can be seen from the HFP results, the increase in the peak fuel enthalpy and fuel centerline temperature due to TCD effects are driven by the [ ]^{a,c}. The HFP results also show that with TCD included, the peak fuel enthalpy and no fuel melt criteria are still met.

For the PCMI limiting case, the initial conditions are not affected by TCD since the transient is initiated from isothermal HZP conditions. As previously stated, the nuclear power transient and the amount of fission energy deposited into the fuel already included the effects of TCD, therefore the impact on the fuel temperature and enthalpy increase is expected to be minor. From the HZP results in Table 1, the change in the enthalpy rise when TCD effects are included is approximately [ ]^{a,c}. This change in enthalpy rise is more than offset by the margin calculated for the CRR to the criterion given in SRP 4.2 Revision 3. The peak fuel centerline temperature also remains well below the fuel melt temperature, satisfying the no fuel melt criterion.

The above results quantitatively demonstrate the conclusions previously discussed in letter LTR-NRC-12-56-P, specifically that the effects of TCD could be accommodated in the rod ejection analysis without invalidating the conclusions of the Core Reference Report. Furthermore, the results presented in this response demonstrate the credit for peaking factor burndown is NOT required to accommodate TCD and meet the rod ejection accident acceptance criteria defined in the SRP 4.2 Revision 3.

a,c

Figure 1: Core Relative Power and Minimum DNBR vs. Time

Table 1: Rod Ejection Fuel Temperature and Enthalpy Results

Parameters	**Without TCD**	**With TCD**
HFP:		a,c
Initial enthalpy (cal/g)		
Peak enthalpy (cal/g)		
Initial Centerline Temperature (°F)		
Peak Centerline Temperature (°F)		
HZP:		
Peak enthalpy (cal/g)		
Peak Centerline Temperature (°F)		

CRR-020 *(non-LOCA accidents)*

A 3-D kinetics code is used to analyze the AP1000 rod ejection accident. Will the same code be used for the future reload analysis?

Westinghouse Response to CRR-020

The same 3-D kinetics methodology used for the Core Reference Report rod ejection analysis will be used for reload cycles. This methodology is consistent with WCAP-15806-P-A. Benchmark testing discussed in RAI CRR-003 has demonstrated that the 3-D kinetics code used to perform the calculations gives equivalent results to SPNOVA for representative rod ejection transients, since the same solution methodology has been implemented.

For reload cycles, only the limiting 3-D kinetics cases necessary to confirm the design basis criteria identified in NUREG-0800 Standard Review Plan (SRP) 4.2 Revision 3 for Pellet-Clad Mechanical Interaction, High Clad Temperature, and Core Coolability will be performed. The limiting cases are known based on the analyses performed to support the CRR rod ejection results and conclusions. For each reload cycle, the fraction of fuel exceeding failure limits will be confirmed to be less than the amount assumed in the dose analysis described in CRR Section 15.4.8.3

CRR-021 *(non-LOCA accidents)*

The FIGHT-H code has been used to support transient analysis to provide fuel rod temperature data at different power and burn-up levels. The accuracy of the code affects the calculated Doppler Feedback Effect and the power level for AOOs and rod ejection accident. Evaluate the difference between the FIGHT-H code and the fuel performance code which models the thermal conductivity degradation properly. Based on the evaluation, determine the limitations of the FIGHT-H code and the impact on the calculated DNBR, peak linear power density, transient power level and cladding strain.

Westinghouse Response to CRR-021

The FIGHTH code provides fuel temperatures for the generation of cross-sections and the Doppler feedback for the core simulator code ANC. The FIGHTH calculation of fuel temperatures is best estimate and includes the effects of TCD. FIGHTH fuel temperatures are also used as the basis for the Doppler feedback during the modeling of the rod ejection nuclear power transient as discussed in the response to Question CRR-019. FIGHTH does not provide fuel temperatures as input for any of the safety analyses. Maximum and minimum fuel temperatures for the safety analyses are generated by the PAD code. TCD impacts on the various safety analyses have previously been discussed in letter LTR-NRC-12-56-P.

Best estimate fuel temperatures calculated by FIGHTH and PAD with TCD effects included tend to agree very well. However, during Phase 2 of the Core Reference Report (CRR) audit review meetings a question was raised about specific instances where FIGHTH [ ]^{a,c} relative to PAD with TCD effects. In particular, these differences occur for the **AP1000** PWR fuel type in the burnup range of [ ]^{a,c} and at linear heat rates of [ ]^{a,c}, with a maximum difference of approximately [ ]^{a,c}. This narrow range of differences is expected to have an insignificant effect on the analyses documented in the CRR.

The impacts of the noted differences on the Doppler feedback will be negligible. For the rod ejection transient, the Doppler feedback is dictated by the [ ]^{a,c}. Figure 1 illustrates the node heat generation (node power in kW/ft) and the corresponding node average power for the limiting HFP case presented in the CRR. As seen in Figure 1 the peak linear heat generation rate peaks at approximately [ ]^{a,c} kW/ft, below the level in question discussed above. Therefore, the observed differences in fuel temperature between FIGHTH and PAD with TCD will not have any effect on the rod ejection nuclear power transient. Since the calculation of DNBR, peak fuel temperatures and enthalpy are all based on PAD safety analysis fuel temperatures, there is no impact on these parameters due to fuel temperature differences between FIGHTH and PAD.

The core power response during an AOO is driven by the bounding Doppler power coefficient inputs provided to the safety analysis models. Calculated maximum and minimum Doppler power coefficients versus safety analysis inputs are shown in Figures 2 and 3. In these figures, the data labeled as "ANC930" includes TCD effects and the data labeled "ANC8" does not include TCD effects. As can be seen from these figures, significant margin exists to the maximum and minimum Doppler power

coefficient limits used in the safety analysis models. The differences between the FIGHTH and PAD with TCD fuel temperatures are relatively small, and the largest differences previously mentioned would only occur in a few nodes in the core, if at all. Therefore, any impacts on the calculated Doppler power coefficients would be insignificant and well within the observed margins to the safety analysis limits.

The last consideration of the effects of the fuel temperature differences between FIGHTH and PAD with TCD would be in the analysis of the core power distribution at the limiting AOO overpower conditions. During some AOOs, the core can be in an overpower condition where the peak linear heat rate approaches the limit to preclude fuel centerline melt. At these high local powers, the observed differences in fuel temperature would result in only small differences in local Doppler feedback and hence very small changes in local power. These changes in local power are well within the calculation uncertainties of the hot spot power (F_Q).

Due to these very minor impacts due to differences in fuel temperatures between FIGHTH and PAD with TCD, there are no limitations placed on the current uses of FIGHTH in the nuclear design process.

a,c

Figure 1: Node Power and Fuel Temperature for the HFP Rod Ejection Transient

a,c

Figure 2: Calculated Most Negative Doppler Power Coefficient vs. Design Limit

a,c

Figure 3: Calculated Least Negative Doppler Power Coefficient vs. Design Limit

CRR-022 *(non-LOCA accidents)*

Loss of flow and LOOP events are the most limiting in terms of system pressure with a margin of about 50 psid before the thermal conductivity degradation is considered. Westinghouse did not anticipate that the peak system pressure is going to exceed the limit if the thermal conductivity degradation is properly modeled; however, the technical basis supporting the conclusion has not been presented. Provide quantitative evidence to justify this conclusion.

Westinghouse Response to CRR-022

The two most limiting transients with respect to peak reactor coolant system pressure in the Core Reference Report are the Loss of Load/Turbine Trip (LOL/TT) event and the Locked Rotor event. Both of these transient are potentially impacted by thermal conductivity degradation (TCD).

To quantify the impact of TCD, updated maximum fuel temperatures, calculated with PAD 4.0 +TCD, were used to develop revised LOFTRAN fuel conductivity parameters. The temperatures used bound fuel properties up to a fuel burn-up of 62000 MWD/MTU. The results of the TCD sensitivity on the Locked Rotor analysis are presented in Table 1 and the results of the TCD sensitivity on the Loss of Load / Turbine Trip event are presented in Table 2. As can be seen, the impact of TCD on the peak RCS pressure is very small (< 5 psid) compared to the margin available in the analyses.

Table 1: Locked Rotor Peak RCS Pressure Comparison considering TCD Effects

Parameter	TCD Sensitivity	Core Reference Report Analysis
RCS Pressure Acceptance Criterion	2748.5 psia	
Peak RCS Pressure at the Reactor Coolant Pump Discharge	2719.82 psia	2716.30 psia
Time of Peak RCS pressure	3.4 seconds	3.4 seconds

Table 2: LOL/TT Maximum RCS Pressure Comparison considering TCD Effects

Parameter	TCD Sensitivity	Core Reference Report Analysis
RCS Pressure Acceptance Criterion	2748.5 psia	
Peak RCS Pressure at the Reactor Coolant Pump Discharge	2728.44 psia	2727.32 psia
Time of Peak RCS pressure (sec.)	9.000	8.900

RAI CRR-024 *(Fuel Seismic)*

What is the basis for assuming the applicability of the Westinghouse damping test data from older fuel assembly designs to the AP1000 Core Reference Report fuel assembly design?

Westinghouse Response to CRR-024

The test assembly used in the Westinghouse fuel assembly damping test is a PWR fuel assembly [ ]^{a,c}. Although the array size, number of thimbles, and number of mid grids vary between fuel assembly designs, the basic structure of this test assembly is similar to all other PWR fuel assemblies including the **AP1000** PWR fuel assembly. This assembly was tested in air, still water, and flowing water using the same test setup.

Other fuel vendors have also performed fuel assembly damping tests with similar but not identical PWR fuel assemblies to the Westinghouse fuel assembly. All of these fuel assemblies have geometric similarities in that they are comprised of a square array of fuel rods with guide tubes and spacer grids. A comparison of the published test data from these other manufacturers is described in the following paragraphs and demonstrates that the damping value of all PWR fuel assemblies is similar. The main reason why fuel assembly damping coefficients for different designs are similar is based on this geometric similarity. Additionally, Westinghouse currently uses the same damping coefficient for all types of Westinghouse fuel (in Westinghouse type reactors) for seismic analysis.

The Westinghouse test data provides a direct comparison of the damping values due to the different environments (air, still water, and flowing water) as shown in Figures 1 and 2. Furthermore, the data in Figure 2 and the data given in Reference 2 provide similar damping values. Other similarities between the data are (1) the damping values in air and still water are amplitude dependant and (2) the damping in flowing water is significantly higher than in air and still water and is less amplitude dependant. This comparison demonstrates that minor differences in design do not result in significant differences in damping. It should also be noted that the test assembly used in Reference 2 has a 17x17 array, 8 mid grids and 264 fuel rods with 0.374 inch OD. These parameters are the same as for the **AP1000** PWR fuel assembly.

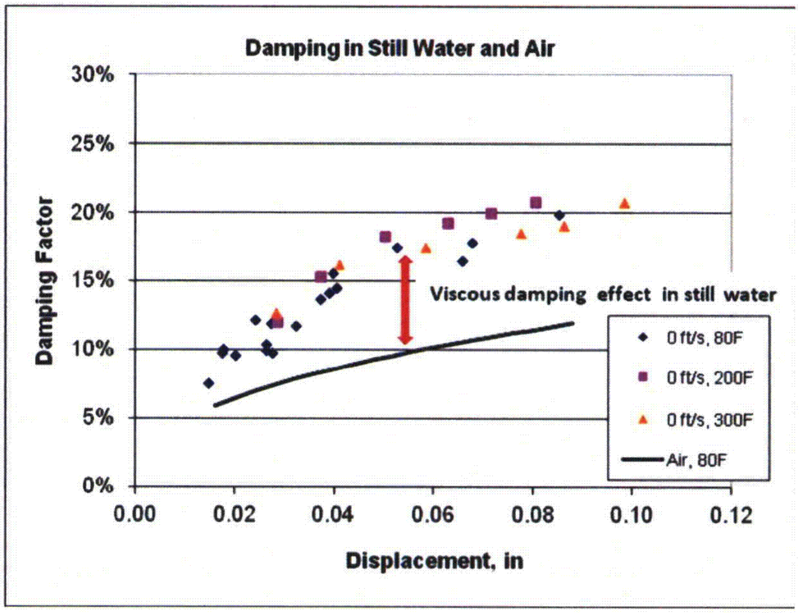

Figure 1: Fuel Assembly Damping Factors in Air and Still Water – Westinghouse Test Data

a,c

Figure 2: Fuel Assembly Damping Factors in Still and Flowing Water – Westinghouse Test Data

Additional comparisons of damping data from different fuel assembly designs are provided in Table 1. Both the French at Cadarache (Reference 2) and B&W (Reference 3) have performed damping tests in flowing water. Both of these tests were performed using fuel assemblies with a 17x17 array and 264 fuel rods. This is the same as the **AP1000** PWR fuel assembly. These results from different fuel vendors and different tests show that PWR fuel assembly damping values in flowing water are very similar and that the results are not fuel design dependent within the range of PWR fuel designs tested.

Table 1*: Summary of Fuel Assembly Damping Data

a,c

[ ]

Summary

All PWR fuel assemblies have similar parameters and structures, consisting of thimble tubes, spacer grids and fuel rods. Fuel assembly damping in flowing water mainly depends on the hydraulic forces acting on the fuel rods and grid straps, which directly correlates to axial flow velocity. The test results in Table 1 show similar damping values under similar axial flow conditions. Therefore, fuel assembly damping in flowing water is not fuel design dependent for PWR fuel. Westinghouse damping data from tested fuel assembly designs can be applied to all PWR fuel designs, including the **AP1000** PWR fuel assembly design.

References

1. R. Y. Lu and D. D. Seel, Westinghouse USA, "PWR Fuel Assembly Damping Characteristics," Proceedings of ICONE 14, 14th International Conference on Nuclear Engineering, July 17-20, 2006, Miami, Florida, USA.

2. S. Pisapia, et al. "Modal Testing and Identification of a PWR Fuel Assembly," Transactions of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17), Paper #C01-4, Prague, Czech Republic, August 17-22, 2003.

3. F. E. Stokes and R. A. King, "PWR Fuel Assembly Dynamic Characteristics," International Conference on Vibration in Nuclear Power Plants, Keswick, United Kingdom, May 9-12, 1978 (BNES).

CRR-025 *(Fuel Seismic)*

Provide justification for the applicability of crush test data/analysis from WCAP-12488-P-A Addendum 1 Rev. 1 to the eMidGrids and eIFMs used in the AP1000 Core Reference Report fuel assembly design.

Westinghouse Response to CRR-025

The EOL seismic/LOCA evaluation of the **AP1000** PWR fuel assembly will not be based on crush test data from Addendum 1 to WCAP-12488 (Reference 1). Instead, crush strength and through grid stiffness data from tests of **AP1000** PWR specific grids will be used for the EOL evaluation. These tests have been performed using **AP1000** PWR mid grids with gaps between the grid springs and the fuel rods. Gap formation between the grid springs and the fuel rods due to irradiation induced spring relaxation, grid growth, and cladding creep-down is the primary contributor to the reduction in crush strength that occurs at EOL conditions.

The average gap used in the **AP1000** PWR crush tests conservatively exceeds the upper bound average gap based on PIE measurements of grid cell size and fuel rod diameter as shown in Figure 1 below. These PIE measurements are from fuel assemblies with burn-ups comparable to the **AP1000** PWR EOL burn-ups and with fuel rods with the same diameter and material as the **AP1000** PWR fuel rods. These fuel assemblies have RFA style grids. The RFA design is the basis for the **AP1000** PWR mid grid design. The RFA grids have the same material (ZIRLO® High Performance Fuel Cladding Material) and strap thicknesses as the **AP1000** PWR grids. The RFA mid grid has a diagonal inner strap spring and a vertical outer strap spring, which are very similar to the **AP1000** PWR springs. Although the **AP1000** PWR grid is taller than the RFA grids and has a longer contact length spring and dimple. Neither of these features are expected to affect the EOL gap.

a,c

Figure 1: Grid Spring to Fuel Rod Gaps at EOL Conditions from PIE Measurements

The EOL grid crush strength from these **AP1000** PWR tests is [ ]^{a,c} Compared to the BOL crush strength of [ ]^{a,c}, the EOL crush strength is [ ]^{a,c} less. The EOL through grid stiffness from these tests is [ ]^{a,c}. Compared to the BOL stiffness of [ ]^{a,c}, the EOL stiffness is [ ]^{a,c} less.

These results are based on crush tests of the mid grid which is the grid with the highest seismic/LOCA loading. Additional crush testing of the **AP1000** PWR IFM grids at EOL conditions is not necessary because the IFM grids have a [ ]^{a,c} nominal gap between the fuel rod and the grid dimples at BOL conditions. As such, the difference between the BOL and the EOL crush strength is expected to be minimal. [ ]^{a,c}

References

1. WCAP-12488-A, Addendum 1-A, Rev. 1, "Addendum 1 to WCAP-12488-A Revision to Design Criteria." Westinghouse Electric Company, January 2002.

CRR-026 *(non-LOCA accidents)*

An accurate or a conservative prediction of core initial stored energy is important to the containment peak pressure calculation for a postulated large break loss-of-coolant accident. The core initial stored energy used as input to this calculation is claimed to be the highest at zero power level and BOC [ ]^{a,c} condition in contrast to full power level and EOC [ ]^{a,c} condition. Describe the core initial stored energy calculation method and explain how the core at 0% rated power can be the most limiting in regards to initial core stored energy.

Westinghouse Response to CRR-026

Core stored energy (CSE) is defined as the amount of energy in the fuel rods in the core above the local coolant temperature. The Fuel Rod Design (FRD) group calculates a normalized CSE for Containment and Radiological Analysis (CRA) using the following equation:

a,c

[ ]

The CSE is calculated using a simplified, conservative approach by dividing by the local linear power (q'), which normalizes the CSE. The normalization of the local stored energy with respect to the local linear power, as given in the above formula, ensures that the variation of the CSE with power is small. It also accounts for the improved heat transfer capability of the fuel at higher power levels.

The units of normalized CSE are in full power seconds (fps). It can be considered as a thermal time constant, measuring the rate at which heat decays from the fuel to the coolant. The standard thermal time constant equation is listed below:

$$\tau = \frac{\rho\, C_p\, V}{U\, A}$$

where:

τ	=	time constant, hr
ρ	=	density, kg/m^3
C_p	=	specific heat, J/kg-K
V	=	volume, m^3
U	=	overall heat transfer coefficient, W/m^2-K
A	=	area, m^2

[

]a,c

Thus, the CSE is limiting at [ ]a,c It should be noted that the CSE is limiting at [ ]a,c not zero power, as postulated in the question above.

a,c

[ ]

Figure 1: AP1000 PWR Normalized Core Stored Energy as a Function of Local Power

Normalized CSE is calculated using [ ]a,c generated using the current Performance Analysis and Design models (PAD 4.0) based on a peak rod operating at limiting conditions. However, CSE is a whole-core parameter, and [ ]a,c Maximum fuel average temperatures (plus uncertainties) used to calculate normalized CSE occur near [ ]a,c as seen in Figure 2.

However, the primary impact of Thermal Conductivity Degradation (TCD) is increased fuel temperature as burnup is accumulated. [ ]^{a,c} as shown below. While the TCD model has not been approved by the Nuclear Regulatory Commission (NRC), it has been reviewed and validated against other fuel performance codes.

[ ]^{a,c}

a,c

Figure 2: **AP1000** PWR Maximum Fuel Average Temperature (Plus Uncertainties) as a Function of Rod Average Burnup

As seen in Figure 3, [ ]a,c remains limiting for normalized CSE when accounting for TCD.

a,c

[ ]

Figure 3: **AP1000** PWR Normalized Core Stored Energy as a Function of Local Power with Thermal Conductivity Degradation

[

]^{a,c} as seen in Figures 4 – 6.

a,c

Figure 4: **AP1000** PWR Normalized Core Stored Energy as a Function of Rod Average Burnup, [ ]^{a,c} Local Power

a,c

Figure 5: **AP1000** PWR Normalized Core Stored Energy as a Function of Rod Average Burnup, [ ]a,c Local Power

a,c

Figure 6: **AP1000** PWR Normalized Core Stored Energy as a Function of Rod Average Burnup, [ ][a,c] Local Power

Thus, the normalized CSE is limiting at [ ][a,c] at the time in life that corresponds to [ ][a,c] For PAD 4.0, the peak normalized CSE occurs at [ ][a,c] When modeling the impacts of TCD, the peak CSE occurs at [ ][a,c]

This is demonstrated in Figures 1 – 6.

[ ][a,c]

[

]a,c

Section F

Westinghouse

Westinghouse Electric Company
Nuclear Services
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: greshaja@westinghouse.com

LTR-NRC-13-3

January 10, 2013

Subject: Second Transmittal of Westinghouse Responses to NRC RAIs on WCAP-17524, "AP1000 Core Reference Report" (Proprietary/Non-Proprietary).

Enclosed are copies of the proprietary and non-proprietary versions of the responses to the NRC RAIs for WCAP-17524, "AP1000 Core Reference Report" dated March 2012.

Also enclosed is:

1. One (1) copy of the Application for Withholding Proprietary Information from Public Disclosure, AW-13-3586 (Non-Proprietary), with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference AW-13-3586 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J.A. Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

Westinghouse Electric Company
Nuclear Services
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: greshaja@westinghouse.com

AW-13-3586

January 10, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-13-3 P-Attachment, "Second Transmittal of Westinghouse Responses to NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report'" (Proprietary)

Reference: Letter from J. A. Gresham to Document Control Desk, LTR-NRC-13-3, dated January 10, 2013

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-13-3586 accompanies this Application for Withholding Proprietary Information from Public Disclosure, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject 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 proprietary aspects of the application for withholding or the accompanying affidavit should reference AW-13-3586 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, 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:

James A. Gresham, Manager
Regulatory Compliance

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

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, in Nuclear Services, 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 LTR-NRC-13-3 P-Attachment, "Second Transmittal of Westinghouse Responses to NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report'" (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter, LTR-NRC-13-3, and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the review of WCAP-17524, and may be used only for that purpose.

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

(a) Obtain NRC approval of the **AP1000**® Pressurized Water Reactor (PWR) Advanced First Core, as documented in WCAP-17524, "AP1000 Core Reference Report".

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 assisting customers in obtaining license changes for the **AP1000** PWR.

(b) This document establishes a portion of the licensing basis for the **AP1000** PWR.

(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 technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

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 for generic and/or plant-specific review and approval.

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

Second Transmittal of Westinghouse Responses to NRC RAIs on WCAP-17524, "AP1000 Core Reference Report" (Non-Proprietary)

January 2013

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

CRR-001 *(spent fuel pool criticality)*

With the movement to the Advanced First Core (AFC), there is a change from a 3-region design to a 5-region design. With respect to spent fuel pool criticality analyses as described in APP-GW-GLR-029, Revision 3, titled, "AP1000 Spent Fuel Storage Racks Criticality Analysis,"

a) How does the change to a 5-region core affect the previously identified limiting fuel assembly depletion characteristics? For example, the current analysis in APP-GW-GLR-029 identifies the limiting assembly insert combination during fuel depletion as having [

]^{a,c}

b) Considering the AFC and future cycle core designs, how is the limiting assembly insert combination affected?

c) Also, with a change in the core design, it is likely that the axial burnup distributions have also changed. Demonstrate that the change in core design, including the effect of reload core designs, either does not affect the limiting axial burnup distributions as discussed in APP-GW-GLR-029 or update the safety analyses in APP-GW-GLR-029 to include the appropriate distributions and analysis impacts.

d) Demonstrate that the cumulative impact of the AFC and reload core designs still satisfy the appropriate criteria in 10 CFR 50.68.

Westinghouse Response to CRR-001

a) The spent fuel pool (SFP) criticality analysis described in APP-GW-GLR-029P Revision 3 considered the Advanced First Core (AFC) design 5-region core, as well as the original 3-region first core design and anticipated reload cycles, when determining the limiting fuel assembly types for depletion characteristics. This is evidenced by the fact that both WABA and Pyrex burnable absorber designs were considered in combination with the IFBA burnable absorber design in Table 5.14 of the subject analysis. The WABA/IFBA combination is used in the AFC design and the PYREX/IFBA combination is used in the original 3-region first core design.

   The SFP criticality analysis analyzed conservative combinations for the number of WABA and IFBA rods used in the same assembly. The limiting fuel assembly type identified in the Region 2 SFP criticality analysis used [
   ]^{a,c} The AFC design will actually employ fuel assemblies with maximum burnable absorber combinations of [ ]^{a,c} By bounding the total amount of initial burnable absorber material used in the first cycle (i.e., as quantified by the total Boron-10 initially contained within both the IFBA and WABA rods), the reactivity of the fuel at the time of its final discharge into Region 2 of the SFP is increased, due to variations in the plutonium isotopic content of the discharged fuel.

In addition, both the silver-indium-cadmium and tungsten gray rod designs were considered in the depletion calculations performed to support the Region 2 SFP criticality analysis, as evidenced in Table 5.14 and Appendix F of the subject analysis.

Therefore the implementation of the AFC design and the tungsten gray rod design do not affect the selection of the most limiting fuel type identified in the Region 2 SFP criticality analysis.

b) As discussed above, the depletion calculations performed to support APP-GW-GLR-029P Revision 3 considered the AFC design and typical reload cycles subsequent to the AFC design. This included the potential placement combinations of the most limiting fuel type during its second and third cycle of operation. The selected limiting assembly insert combination has already taken into account the AFC design and subsequent reload cycles.

c) The axial burnup distributions considered in APP-GW-GLR-029P Revision 3 were developed based on both rodded and un-rodded operation of the AFC design and subsequent reload cycles, including both Cycle 2 and a representative equilibrium cycle reload design. The most limiting axial burnup distributions were determined by comparing the relative burnups near the top of the fuel assembly as described in Section 7.2.2 and Appendix E of APP-GW-GLR-029P Revision 3.

The most limiting axial burnup shape used in the SFP criticality analysis is provided in Table 5.11 of APP-GW-GLR-029P Revision 3. This limiting axial burnup shape came from an end of Cycle 1 fuel assembly, taken from the AFC design, which was located directly under a partially inserted RCCA for the entire cycle. The average burnup of this assembly was relatively low at the end of Cycle 1, but the relative axial shape of the burnup distribution is the most limiting, due to the low exposure in the top of the fuel assembly relative to the assembly average. This limiting axial burnup shape was applied to all burnup credit calculations by scaling the relative burnup shape up to higher assembly average burnup values. This approach is very conservative because, in reality, the axial burnup distributions from fully discharged fuel operating over multiple cycles in different core locations would tend to have higher relative burnups near the top of the fuel than are simulated with the scaled axial burnup shape taken from the partially rodded location at the end of Cycle 1. This is demonstrated graphically by the plot of "Weighted Relative Burnup as a Function of Distance from the Top" for fuel from different reload cycles, which is shown on Page E-10 of APP-GW-GLR-029P Revision 3.

Since the limiting axial burnup shape was developed in a conservative manner with appropriate consideration given to both the AFC design and limiting rodded operation, the results of the Region 2 SFP analysis in APP-GW-GLR-029P Revision 3 remain unaffected.

d) As discussed in the above response to CRR-001, the SFP criticality analysis performed in APP-GW-GLR-029P Revision 3 considered all aspects the AFC design 5-region core, as well as the original 3-region first core design and anticipated reload cycles. Therefore the results presented in APP-GW-GLR-029P Revision 3 remain valid and still satisfy the regulatory criteria in 10CFR 50.68.

While the analysis documented in APP-GW-GLR-029P Rev. 3, envelopes both the core and fuel design associated with the Advanced First Core (AFC), a minor discrepancy in the analysis was identified and entered into the Westinghouse Corrective Action Process. Specifically, it was noted that the diameter of the GRCA absorber assumed in the analysis is not consistent with the final design of the GRCAs used in the new core design (i.e., the actual absorber diameter is slightly larger than assumed in the analysis). A preliminary evaluation indicates that the impact of this discrepancy is not significant and would have no impact on the final conclusions of the analysis.

The revised analysis will be available to support a formal resolution of this issue by January 31, 2013.

CRR-004 *(LBLOCA Analysis)*

In the large-break LOCA analysis (Calculation Note APP-SSAR-GSC-772, Rev. 0, "Evaluation of TCD for AP1000 Advanced First Core Application Program and DCD Rev. 19 Best-Estimate LBLOCA ASTRUM Analyses,") using the ASTRUM method to evaluate the impact of the fuel thermal conductivity degradation (TCD), the average fuel assembly burnups are limited to [ ]^{a,c}.

Describe the processes that the average assembly peaking factors and burnups are calculated so as not to underestimate the initial stored energy of the average fuel assemblies. Provide justifications of limiting the burnups of the average assemblies to [ ]^{a,c}.

Westinghouse Response to CRR-004

The average assembly radial peaking is selected to preserve the total core power given the peaking factors of the other rod groups. The average assembly burnup is calculated as follows:

[ ]^{a,c}

Where: Sampled Time-in-Cycle ranges from [ ]^{a,c} for the **AP1000**® plant design.

This results in an analyzed average rod burnup range of approximately [ ]^{a,c}. The expected burnup range of the average rods is [ ]^{a,c}. The difference in the end of cycle burnup results in a maximum average fuel temperature difference of about [ ]^{a,c} at the peak power elevation in the average rods (the difference is slightly less at lower power elevations).

Sensitivity studies were executed for two of the TCD analysis cases, run031 (the limiting ASTRUM case prior to the TCD analysis) and run069 (the limiting ASTRUM case from the TCD analysis), to determine the impact of increased fuel temperature in the average rods. The average rod burnup for these studies was set to [ ]^{a,c}, which results in a higher than expected average rod maximum average fuel temperature.

The average fuel temperature for the average rods at steady-state conditions for both sensitivity studies are shown in Figures 1A and 1B. As expected, the average rod fuel temperatures increased in the sensitivity studies at the increased burnup. This leads to an increase in the average rod peak cladding temperature (PCT) during the Large Break LOCA (LBLOCA) transient as shown in Figures 2A and 2B. However, it is observed that the increased temperature in the average rods does not substantially impact the global response and had only a small impact on the calculated hot rod PCTs (Figures 3A and 3B). As such, it is concluded that the modeling approach for the average rod burnup is acceptable for the **AP1000** plant TCD analysis.

a,c

[ ]

Figure 1A: Average Fuel Temperature in the Average Rods for ASTRUM Case 031 Sensitivity Study

a,c

[ ]

Figure 1B: Average Fuel Temperature in the Average Rods for ASTRUM Case 069 Sensitivity Study

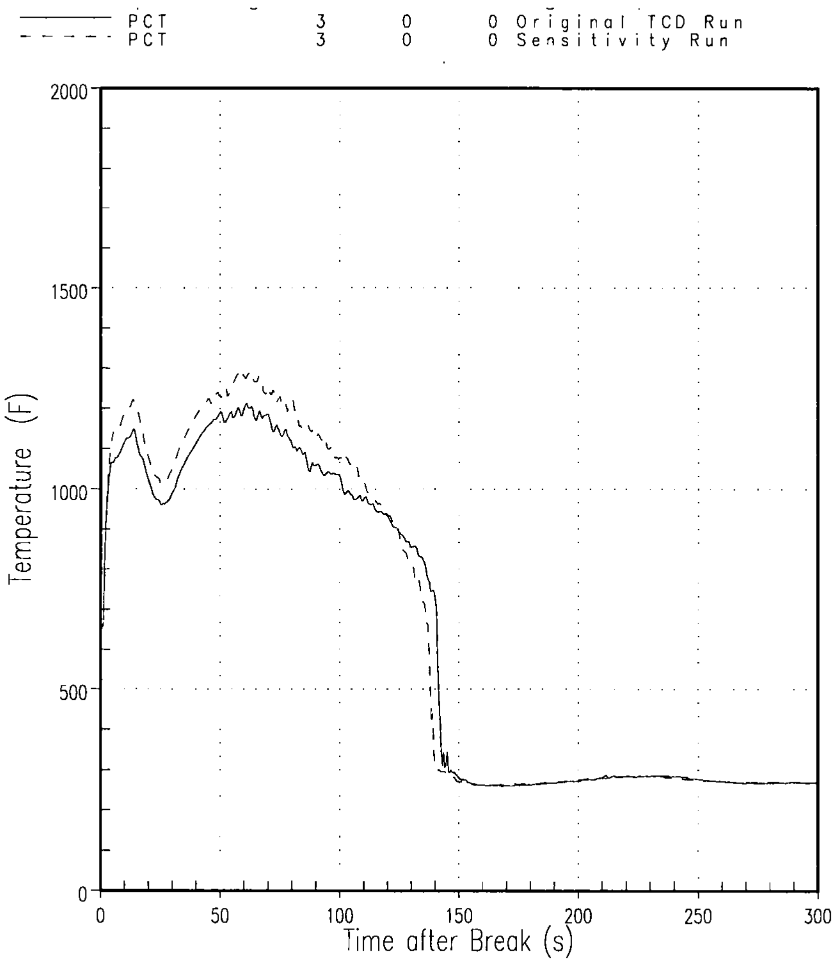

Figure 2A: Average Rod Peak Cladding Temperature for ASTRUM Case 031 Sensitivity Study

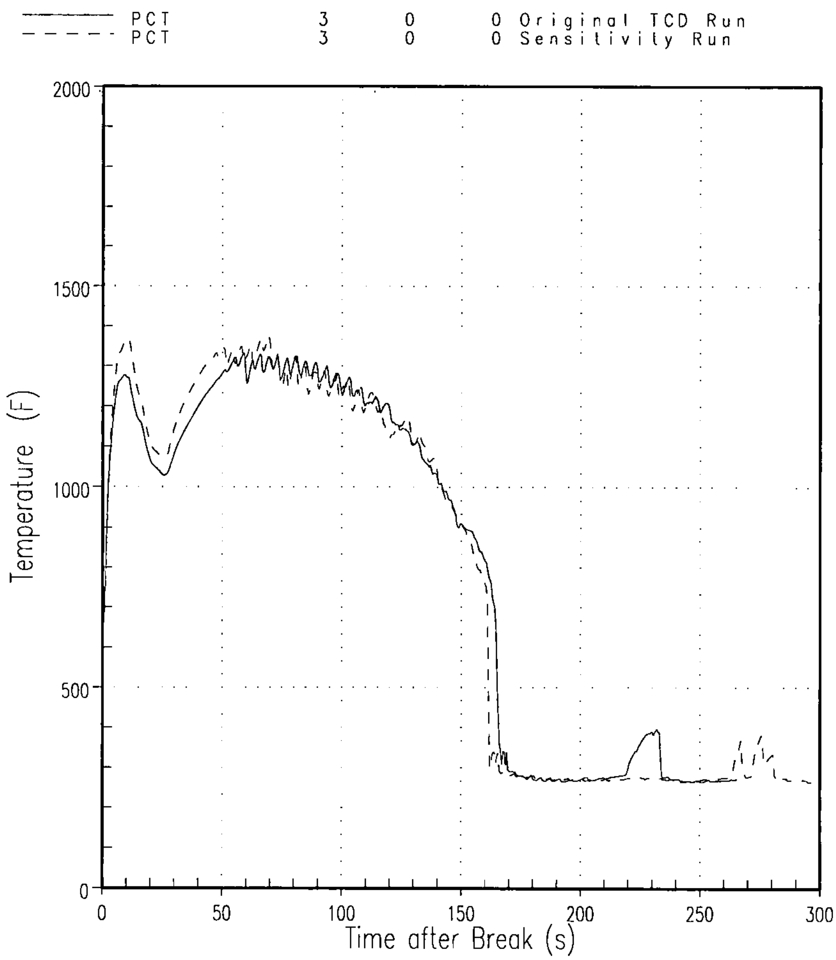

Figure 2B: Average Rod Peak Cladding Temperature for ASTRUM Case 069 Sensitivity Study

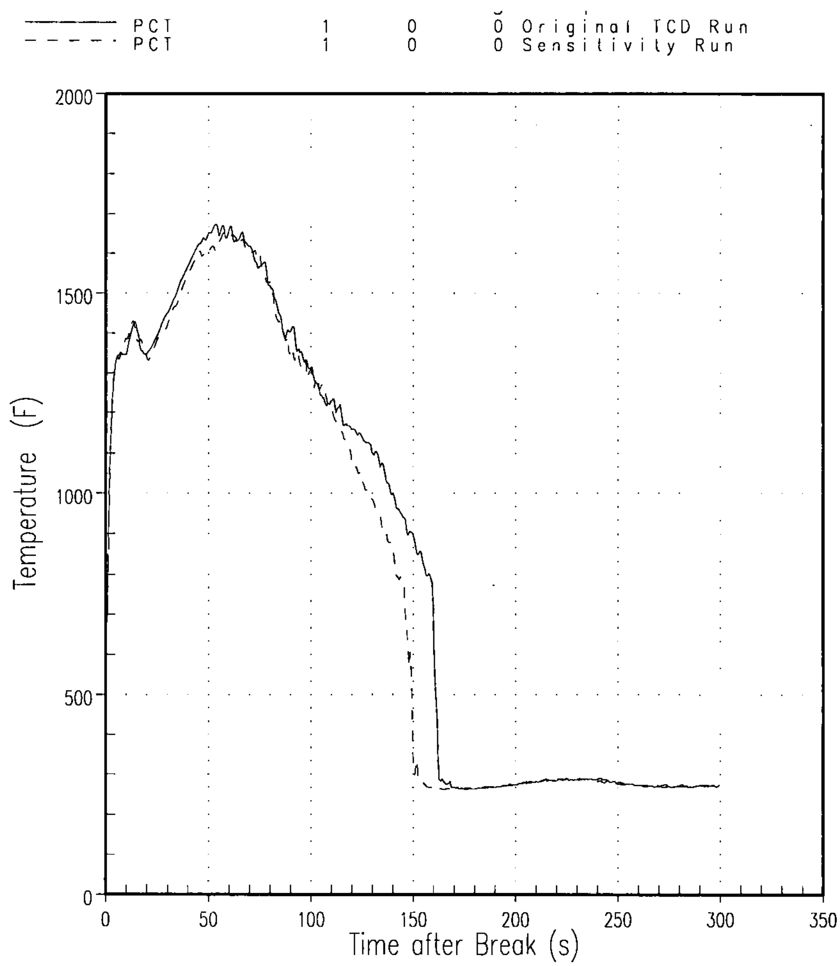

Figure 3A: Hot Rod Peak Cladding Temperature for ASTRUM Case 031 Sensitivity Study

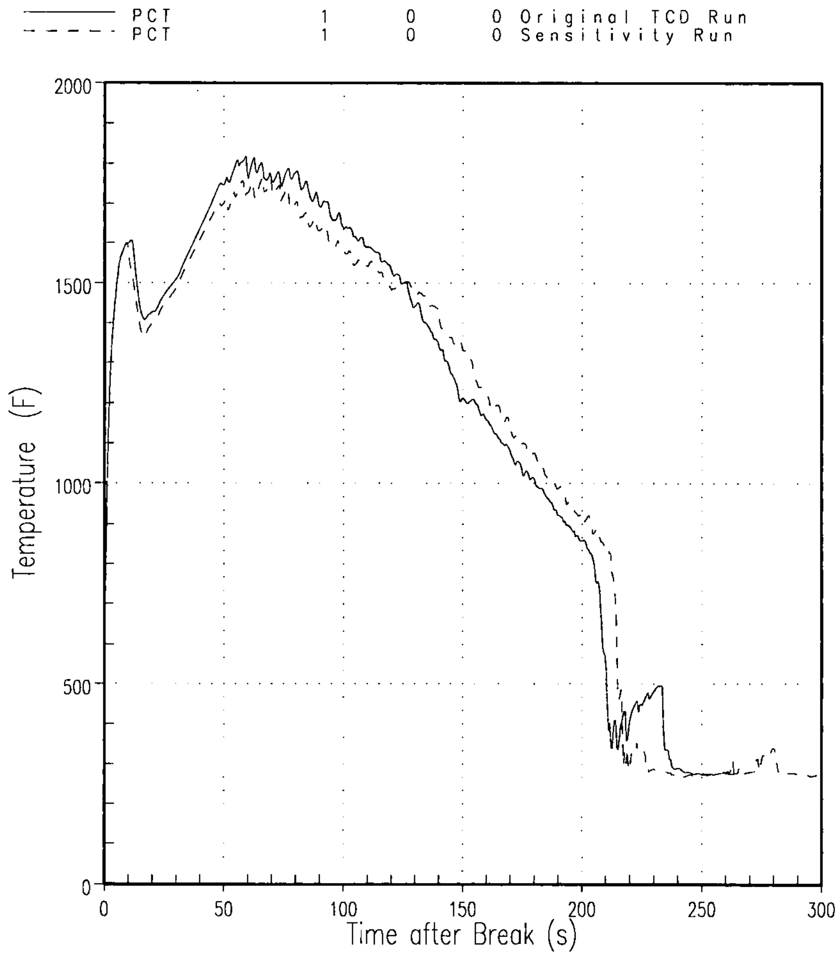

Figure 3B: Hot Rod Peak Cladding Temperature for ASTRUM Case 069 Sensitivity Study

CRR-016 *(Calc Notes-Open Items)*

During the audit held Aug 14-16, 2012, it was noted that some of the calc notes used to support WCAP-17524-P contain open items. The submitted topical report must be based on final analyses following approved quality assurance programs. Therefore, close all related open items and update WCAP-17524-P if necessary.

Westinghouse Response to CRR-016

Westinghouse has identified all open items in the calculation notes that support the Core Reference Report and has classified them into the following three categories:

1) Administrative/Editorial Corrections- These particular open items are administrative items that were identified in the calc notes that were reviewed during the Core Reference Report (CRR). These particular items will be resolved and the open items will be closed.

2) Parameter verification open items will be updated in the configured database. This will enable approximately 32 open items to be verified and closed.

3) Validate inputs that have been used in the updated safety analysis calculations are still the appropriate values and are still valid. If parameters were revised for any reason, an evaluation will be performed to demonstrate that the conclusions for updated safety analysis are still valid.

Several open items are longer lead type subjects (i.e. 4-loop mixing data) that will take several months to complete all the work.

Westinghouse intends to disposition a large portion of those open items that were identified during the NRC audit with the remainder to be dispositioned as expeditiously as possible to support the issuance of the Core Reference Report SER for June 2013.

Formal closure of all open items will occur in conjunction with revised calculations per our current **AP1000** plant Quality Plan.

CRR-018 *(non-LOCA accidents)*

As part of this topical report, Westinghouse developed a new procedure to mitigate the vessel head vent opening event by discharging high temperature and high pressure primary coolant into the in-containment RWST. Similar approaches have been developed for loss of feedwater heater event, inadvertent opening of CMT valves and CVCS malfunction. It was indicated by Westinghouse that the discharge of primary coolant into the containment will cause high containment air space temperature. Demonstrate that the in-containment equipment qualification program has already taken into account these containment heat up events. And the maximum allowable containment air space temperature for all the equipments required for operation is higher than the maximum containment air space temperature during these AOO events.

Westinghouse Response to CRR-018

The equipment qualification program methodology is described in DCD appendix 3D. The qualification of a safety-related component is based on an aging program followed by qualification for seismic, high temperature, high pressure, high radiation, and post accident chemistry conditions. The aging program includes the effect of temperature for normal plant operation plus 72 hours of accumulated Abnormal Group 1 conditions and 24 hours of accumulated Abnormal Group 2 conditions. The Abnormal Group 1 qualification temperature is 150°F. The Abnormal Group 2 qualification temperature is 250°F. Events such as a core makeup tank inadvertent injection and a CVS malfunction are considered to be Group 1. Long term PRHR operation or a small loss of coolant accident is considered to be Group 2. If these less severe accidents occur, the safety-related components are still within their aging basis, and the components are qualified for the more severe Condition IV LOCA or main steam line break because the qualification program includes aging for the components that undergo full Condition IV accident qualification.

There are no specific containment temperature/pressure analyses for all the Abnormal Group 1 and 2 design basis events. It is acknowledged that long term operation of the passive residual heat removal and mass input to the IRWST from the reactor vessel vent pipe will eventually cause an increase in containment temperature, containment pressure, and potentially some IRWST overflow. Most of the reason for the increase is the refueling water boiloff due to decay heat and not the head vent mass flow. For long term passive heat removal from the PRHR heat exchanger to the refueling water, a containment temperature of 230°F is calculated. This is less than the bounding 250°F basis for and Abnormal Group 2 event. The small LOCA containment temperature is less than 230°F. Therefore, a safety-related component in containment would be still qualified for additional Abnormal Group 1, Abnormal Group 2, plus the Condition IV limiting design basis event conditions described in the Appendix 3D figures and tables, given the way that the EQ methodology is applied.

Therefore, it is demonstrated that a calculation exists for the accidents that create the Abnormal Group 2 environment, and it has been demonstrated that the qualification methodology accounts for the Abnormal Group 2 events. Opening of core makeup tank valves or a CVS malfunction do not cause a high containment temperature. These events are considered to be Abnormal Group 1. A 150°F temperature is a conservative limit for events that do not release significant additional energy to containment.

As long as these PRHR and/or head vent-actuation events do not exceed a containment temperature of 250°F for an accumulated duration of 24 hours, then the affected components are bounded by component pre-accident aging design basis conditions. On top of the pre-aging conditions, the complete equipment qualification program includes Condition IV (e.g., main steam break or LOCA) seismic, radiation, chemistry, pressure and temperature conditions as defined in the equipment qualification methodology. If the accumulated Abnormal Group 1 and 2 accidents exceed the aforementioned aging bases, Westinghouse would no longer be able to claim that safety-related equipment, including the head vent valves, are qualified, and equipment replacement, additional testing, or further evaluation would be required.

CRR-023 *(LOCA)*

Westinghouse has used the previously approved containment mass and energy methodology to perform containment peak pressure calculation during LOCA. Does the corrected thermal conductivity degradation model affect the peak containment pressure and temperature calculation?

Westinghouse Response to CRR-023

The approved methodology for calculating mass and energy releases from a large break loss-of-coolant accident (LOCA) uses a value for the initial core stored energy (CSE) as an input. The resulting LOCA mass and energy releases are used in the approved methodology for calculating the long-term containment pressure and temperature response. The method for calculating the CSE is given in the request for additional information (RAI) CRR-026. Based on the description of the calculation for CSE in RAI CRR-026, the current value that is used as an input for CSE for generating large break LOCA mass and energy releases remains conservative when the effects of thermal conductivity degradation (TCD) are considered. Therefore, there is no change to peak calculated containment pressure or temperature resulting from a large break LOCA.

CRR-027 *(Fuel Seismic)*

What is the hydraulic damping coefficient that Westinghouse plans to credit for different flow conditions in the fuel assembly seismic response analysis? How would a coast down be handled?

Westinghouse Response to CRR-027

Figure 1 summarizes Westinghouse fuel assembly damping test data in still and flowing water for water temperatures between 100°F and 300°F. These tests are described in Reference 1. An analysis of the data leads to the following observations of the trends associated with credit for damping:

1) Damping in flowing water increases slightly with vibration amplitude.

2) Damping in flowing water decreases slightly with increasing temperature.

3) Flow velocity has a strong effect on damping and results in a significant increase in damping with increasing flow velocity.

These observations are consistent with other test results that are available publicly (References 2 and 3). The considerations for application of hydraulic damping coefficients due to flowing water are discussed further below.

Fuel assembly damping force in flowing water is actually the summation of fuel structural damping in air (due to material and friction damping), viscous damping in still water and hydraulic damping in flowing water as shown in Equation (1). All three damping coefficients are non-linear.

$$F_d = c_s\dot{x} + c_v\dot{x} + c_h\dot{x} \qquad (1)$$

c_s – Structural damping coefficient in air, mainly increasing with amplitude

c_v – Viscous damping in still water, mainly increasing with vibration velocity

c_h – Hydraulic damping in flowing water, mainly increasing with axial flow velocity

<u>**Vibration Amplitude and Flow Rate Dependence**</u>

Both structural damping and viscous damping terms are vibration-amplitude dependent and increase with vibration amplitude. The hydraulic damping term in flowing water is not vibration-amplitude dependent and it dominates the total damping. Typical fuel assembly displacements during a seismic event with grid impact occurrences are much greater than [ ]^{a,c}. Therefore, the damping data from the flowing water tests with vibration displacements greater than [ ]^{a,c} are used to obtain the damping versus flow velocity curve shown in Figure 2.

Temperature Dependence of Flowing Water Damping

As shown in Figure 1, damping is not very sensitive to temperature. For examples [

]^{a,c} This is consistent with the data from Reference 3 where it was concluded that in the range between 70° to 600°F "damping is minimally affected by temperature in water." It is estimated, based on data from Reference 3, that the damping coefficient from 204°C to 316°C (400°F to 600°F) at 5700 liters/min (~12 ft/s) and at 3800 liters/min (~8 ft/s) is reduced by approximately 3 percentage points. [

]^{a,c}

***AP1000* Plant Pump Coast-Down**

[

]^{a,c}

a,b,c

[ ]

Figure 1: Fuel Assembly Damping Factors in Still and Flowing Water – Westinghouse Test Data

a,b,c

[ ]

Figure 2: Fuel Assembly Damping in Flowing Water

a,c

[ ]

Figure 3: RCS Pump Coast-Down

References

1. R.Y. Lu and D.D. Seel, Westinghouse USA, "PWR Fuel Assembly Damping Characteristics," Proceedings of ICONE 14, 14th International Conference on Nuclear Engineering, July 17-20, 2006, Miami, Florida, USA.

2. S. Pisapia, et al. "Modal Testing and Identification of a PWR Fuel Assembly," Transactions of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17), Paper #C01-4, Prague, Czech Republic, August 17-22, 2003.

3. F. E. Stokes and R. A. King, "PWR Fuel Assembly Dynamic Characteristics," International Conference on Vibration in Nuclear Power Plants, Keswick, United Kingdom, May 9-12, 1978 (BNES).

Section G

Westinghouse

Westinghouse Electric Company
Nuclear Services
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: greshaja@westinghouse.com

LTR-NRC-13-18

March 28, 2013

Subject: Westinghouse Response to Supplemental NRC RAIs on WCAP-17524, "AP1000 Core Reference Report" (Proprietary/Non-Proprietary)

Enclosed are copies of the proprietary and non-proprietary versions of, "Westinghouse Response to Supplemental RAIs on WCAP-17524, 'AP1000 Core Reference Report'" dated March 28, 2013.

Also enclosed is:

1. One (1) copy of the Application for Withholding Proprietary Information from Public Disclosure, AW-13-3676 (Non-Proprietary), with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference AW-13-3676 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

Westinghouse

Westinghouse Electric Company
Nuclear Services
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: greshaja@westinghouse.com

AW-13-3676

March 28, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-13-18 P-Attachment, "Westinghouse Response to Supplemental NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report'" (Proprietary)

Reference: Letter from James A. Gresham to Document Control Desk, LTR-NRC-13-18, dated March 28, 2013

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-13-3676 accompanies this Application for Withholding Proprietary Information from Public Disclosure, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject 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 proprietary aspects of the application for withholding or the accompanying affidavit should reference AW-13-3676 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AW-13-3676

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, 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:

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 28th day of March 2013

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, in Nuclear Services, 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 LTR-NRC-13-18 P-Attachment, "Westinghouse Response to Supplemental NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report' (Proprietary)", for submittal to the Commission, being transmitted by Westinghouse letter, LTR-NRC-13-18, and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the review of WCAP-17524, and may be used only for that purpose.

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

(a) Obtain NRC approval of the **AP1000**® Pressurized Water Reactor (PWR) Advanced First Core, as documented in WCAP-17524, "AP1000 Core Reference Report."

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 assisting customers in obtaining license changes for the **AP1000** PWR.

(b) This document established a portion of the licensing basis for the **AP1000** PWR.

(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 technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

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 for generic and/or plant-specific review and approval.

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