Calvert Cliffs Nuclear Power Plant

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

CENG
a joint venture of
Constellation Energy · eDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

NRC 11-013

February 11, 2011

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Supplement to License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

REFERENCE: (a) Letter from Mr. T. E. Trepanier (CCNPP) to Document Control Desk (NRC), dated November 23, 2009, License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

During review of information provided to support Calvert Cliffs transition from Westinghouse nuclear fuel to AREVA nuclear fuel, the Nuclear Regulatory Commission staff has requested supplemental information be provided. The supplemental information is provided in Attachment (1). In addition, to resolve outstanding issues with some of the methodologies used in the evaluation of the transition to AREVA nuclear fuel, we propose the adoption of certain license conditions in Appendix C of Renewed License Nos. DPR-53 and DPR-69 for Calvert Cliffs Units 1 and 2, respectively. The proposed license conditions are contained in Attachment (2). The information in this response does not change the No Significant Hazards Determination previously provided in Reference (a).

Attachment (1) contains information that is proprietary to AREVA, therefore, it is accompanied by an affidavit signed by AREVA, owner of the information (Attachment (3). The affidavit sets forth, with specificity, the considerations listed in 10 CFR 2.390(b)(4). Accordingly, it is requested that the information that is proprietary to AREVA be withheld from public disclosure. The non-proprietary version of Attachment (1) is included as Attachment (4).

A001
NRR

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

Very truly yours,

Thomas E. Trepanier
Plant General Manager

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, Thomas E. Trepanier, being duly sworn, state that I am Plant General Manager - Calvert Cliffs Nuclear Power Plant, LLC (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 11 day of February, 2011.

WITNESS my Hand and Notarial Seal:

Wendy L. Hunter
Notary Public

My Commission Expires:

Wendy L. Hunter
NOTARY PUBLIC
Calvert County, Maryland
My Commission Expires 1/9/2014

1/9/2014
Date

TET/PSF/bjd

Attachment: (1) Proprietary Supplemental Information
(2) License Conditions
(3) AREVA Proprietary Affidavit
(4) Non-Proprietary Supplemental Information

cc: **Without Attachment (1)**

D. V. Pickett, NRC
W. M. Dean, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (2)

LICENSE CONDITIONS

Draft License Conditions for Unit 1

Amendment No.	Additional Conditions	Implementation Date
297	For the Asymmetric Steam Generator Transient analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet temperature distribution and application of local peaking augmentation factors. The revised methodology shall be applied to Calvert Cliffs Unit 1 core reload designs starting with Cycle 20.	This amendment is effective immediately and shall be implemented within 60 days of completion of the Unit 1 2012 refueling outage.
	For the Seized Rotor Event analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet flow distribution. The revised methodology shall be applied to Calvert Cliffs Unit 1 core reload designs starting with Cycle 20.	
	For the Control Element Assembly Ejection analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.11, the cycle-specific hot zero power peak average radial fuel enthalpy is calculated based on a modified power dependent insertion limit with Control Element Assembly Bank 3 assumed to be fully inserted (only in the analysis, not in actual plant operations). This revised methodology shall be applied to Calvert Cliffs Unit 1 core reload designs starting with Cycle 20.	
	The Small Break Loss of Coolant accident performed in accordance with the methodology of Technical Specification 5.6.5.b.9 shall be analyzed using a break spectrum with augmented detail related to break size. This revised methodology shall be applied to Calvert Cliffs Unit 1 core reload designs starting with Cycle 20.	
	Core Operating Limits Report Figures 3.1.6, 3.2.3, and 3.2.5 shall not be	

Amendment No.	Additional Conditions	Implementation Date
	changed without prior NRC review and approval until an NRC-accepted generic, or Calvert Cliffs-specific, basis is developed for analyzing the Control Element Assembly Rod Bank Withdrawal Event, the Control Element Assembly Drop, and the Control Element Assembly Ejection (power level-sensitive transients) at full power conditions only.	
	Approval of the use of S-RELAP5 (Technical Specification 5.6.5.b.8) is restricted to only those safety analyses that confirm acceptable transient performance relative to the specified acceptable fuel design limits. Prior transient specific NRC approval is required to analyze transient performance relative to reactor coolant pressure boundary integrity until NRC approval is obtained for a generic or Calvert Cliffs-specific basis for the use of the methodology in Technical Specification 5.6.5.b.8 to demonstrate reactor coolant pressure boundary integrity.	
	For the RODEX2-based fuel thermal-mechanical design analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.3, Calvert Cliffs Unit 1 core reload designs (starting with Cycle 20) shall satisfy the following criteria: a. Predicted rod internal pressure shall remain below the steady state system pressure. b. The linear heat generation rate fuel centerline melting safety limit shall remain below 21.0 KW/ft.	
	For the Control Element Assembly Ejection analysis, Calvert Cliffs Unit 1 core reloads (starting with Cycle 20) shall satisfy the following criteria: a. Predicted peak radial average fuel enthalpy when calculated in accordance with the methodology of Technical Specification 5.6.5.b.11 shall remain below 200 cal/g.	

Amendment No.	Additional Conditions	Implementation Date
	b. For the purpose of evaluating radiological consequences, should the S-RELAP5 hot spot model predict fuel temperature above incipient centerline melt conditions when calculated in accordance with the methodology of Technical Specification 5.6.5.b.8, a conservative radiological source term (in accordance with Regulatory Guide 1.183, Revision 0) shall be applied to the portion of fuel beyond incipient melt conditions (and combined with existing gap source term), and cladding failure shall be presumed.	
	The approval of the emergency core cooling system evaluation performed in accordance with the methodology of Technical Specification 5.6.5.b.7 shall be valid only for Calvert Cliffs Unit 1, Cycle 20. To remove this condition, Calvert Cliffs shall obtain NRC approval of the analysis of once- and twice-burned fuel for core designs following Unit 1 Cycle 20.	

Draft License Conditions for Unit 2

Amendment No.	Additional Conditions	Implementation Date
273	For the Asymmetric Steam Generator Transient analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet temperature distribution and application of local peaking augmentation factors. The revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.	This amendment is effective immediately and shall be implemented within 60 days of issuance.
	For the Seized Rotor Event analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet flow distribution. The revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.	
	For the Control Element Assembly Ejection analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.11, the cycle-specific hot zero power peak average radial fuel enthalpy is calculated based on a modified power dependent insertion limit with Control Element Assembly Bank 3 assumed to be fully inserted (only in the analysis, not in actual plant operations). This revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.	
	The Small Break Loss of Coolant accident performed in accordance with the methodology of Technical Specification 5.6.5.b.9 shall be analyzed using a break spectrum with augmented detail related to break size. This revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.	
	Core Operating Limits Report Figures 3.1.6, 3.2.3, and 3.2.5 shall not be	

Amendment No.	Additional Conditions	Implementation Date
	changed without prior NRC review and approval until an NRC-accepted generic, or Calvert Cliffs-specific, basis is developed for analyzing the Control Element Assembly Rod Bank Withdrawal Event, the Control Element Assembly Drop, and the Control Element Assembly Ejection (power level-sensitive transients) at full power conditions only. Approval of the use of S-RELAP5 (Technical Specification 5.6.5.b.8) is restricted to only those safety analyses that confirm acceptable transient performance relative to the specified acceptable fuel design limits. Prior transient specific NRC approval is required to analyze transient performance relative to reactor coolant pressure boundary integrity until NRC approval is obtained for a generic or Calvert Cliffs-specific basis for the use of the methodology in Technical Specification 5.6.5.b.8 to demonstrate reactor coolant pressure boundary integrity. For the RODEX2-based fuel thermal-mechanical design analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.3, Calvert Cliffs Unit 2 core reload designs (starting with Cycle 19) shall satisfy the following criteria: a. Predicted rod internal pressure shall remain below the steady state system pressure. b. The linear heat generation rate fuel centerline melting safety limit shall remain below 21.0 KW/ft. For the Control Element Assembly Ejection analysis, Calvert Cliffs Unit 2 core reloads (starting with Cycle 19) shall satisfy the following criteria: a. Predicted peak radial average fuel enthalpy when calculated in accordance with the methodology of Technical Specification 5.6.5.b.11 shall remain below 200 cal/g.	

Amendment No.	Additional Conditions	Implementation Date
	b. For the purpose of evaluating radiological consequences, should the S-RELAP5 hot spot model predict fuel temperature above incipient centerline melt conditions when calculated in accordance with the methodology of Technical Specification 5.6.5.b.8, a conservative radiological source term (in accordance with Regulatory Guide 1.183, Revision 0) shall be applied to the portion of fuel beyond incipient melt conditions (and combined with existing gap source term), and cladding failure shall be presumed.	
	The approval of the emergency core cooling system evaluation performed in accordance with the methodology of Technical Specification 5.6.5.b.7 shall be valid only for Calvert Cliffs Unit 2, Cycle 19. To remove this condition, Calvert Cliffs shall obtain NRC approval of the analysis of once- and twice-burned fuel for core designs following Unit 2 Cycle 19.	

ATTACHMENT (3)

AREVA PROPRIETARY AFFIDAVIT

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the attachment to a letter from T.E. Trepanier (Calvert Cliffs Nuclear Power Plant) to Document Control Desk (NRC) entitled "Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel," numbered NRC 11-013 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in

accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 11th

day of February, 2011.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/14
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2014

ATTACHMENT (4)

NON-PROPRIETARY SUPPLEMENTAL INFORMATION

Supplemental Information 1

Small Break LOCA Break Size Increment Study

To verify that the most limiting SB LOCA peak cladding temperature was determined based on a 0.01 ft^2 break size increment the following sensitivity study was performed.

The cladding temperature for the published limiting case (break size of 0.09 ft^2) turned over nearly coincidently with the safety injection tank (SIT) discharge. The peak cladding temperature occurred [ ] after the start of the SIT discharge. However, SIT discharge did not occur for the next smaller break size (0.08 ft^2). Given this information it could be postulated that the limiting peak cladding temperature could occur at a break size between 0.08 ft^2 and 0.09 ft^2.

To further investigate the issue, AREVA performed additional SB LOCA studies, specifically, at 0.087 ft^2, 0.088 ft^2, and 0.089 ft^2. These break areas represent an increment in break diameter of 0.02 inches. Note the SIT injection rate was multiplied by 2 to demonstrate the timing dependence of SIT injection on the hot spot cladding temperature using a single y-axis scale.

The results of those cases are shown below in Figures 1-1 through 1-4. They produced lower peak cladding temperatures than the previously identified limiting case. The results are reasonable, with the time between peak cladding temperature occurrence and SIT discharge monotonically decreasing as break size increased – until, for the limiting case, the SIT discharge occurred almost simultaneously with peak cladding temperature. From the study, it can be seen that the cases performed encompassed the possible timing dependencies between peak cladding temperature and SIT injection. The scope of cases cover the phenomena where the escalation of cladding temperature was terminated by [ ] to the limiting case where the [ ] occurred essentially at the same time. For break sizes less than 0.09 ft^2 (4.1 inches), the model was biased to ensure [ ]. For break sizes larger than 0.09 ft^2, [ ] with an associated reduction in peak cladding temperature. Thus, the supporting calculations confirm that within a change in break diameter of 0.02 inches, the 0.09 ft^2 break remains the limiting case.

[ ]

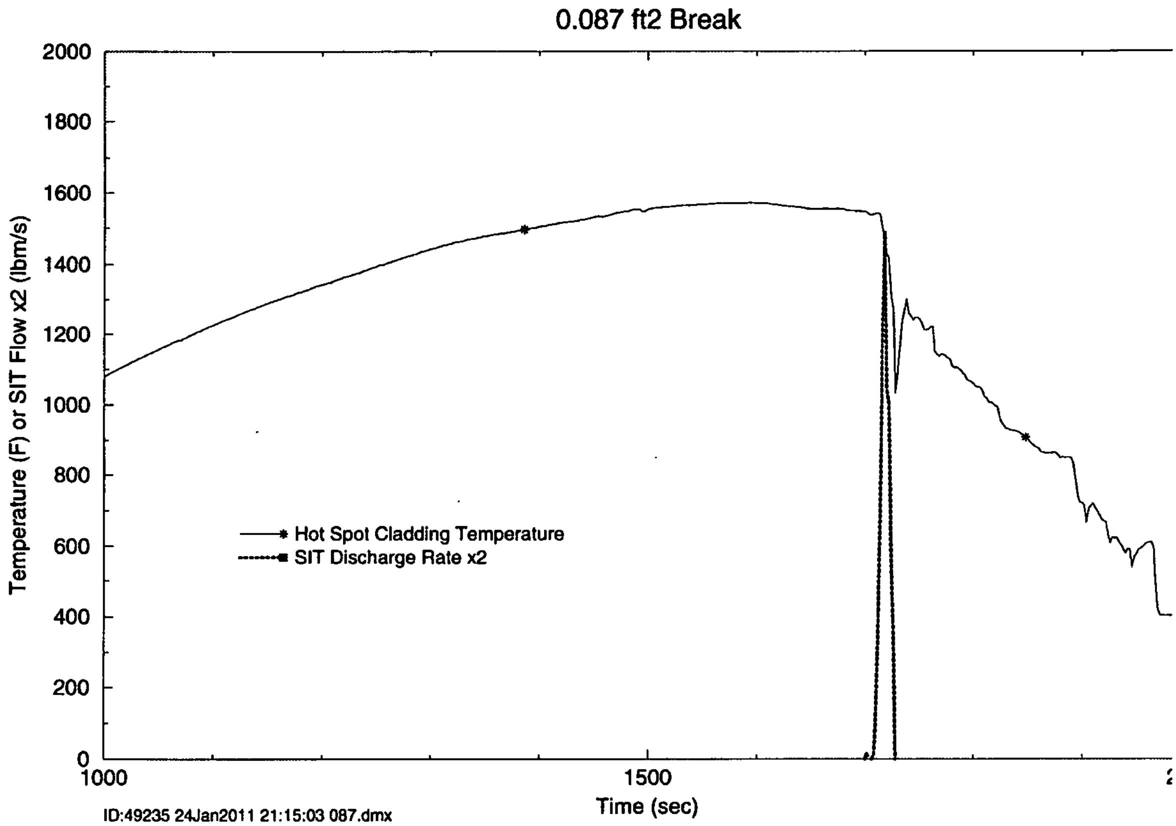


Figure 1-1, Peak Cladding Temperature for a 0.087 ft^2 Break

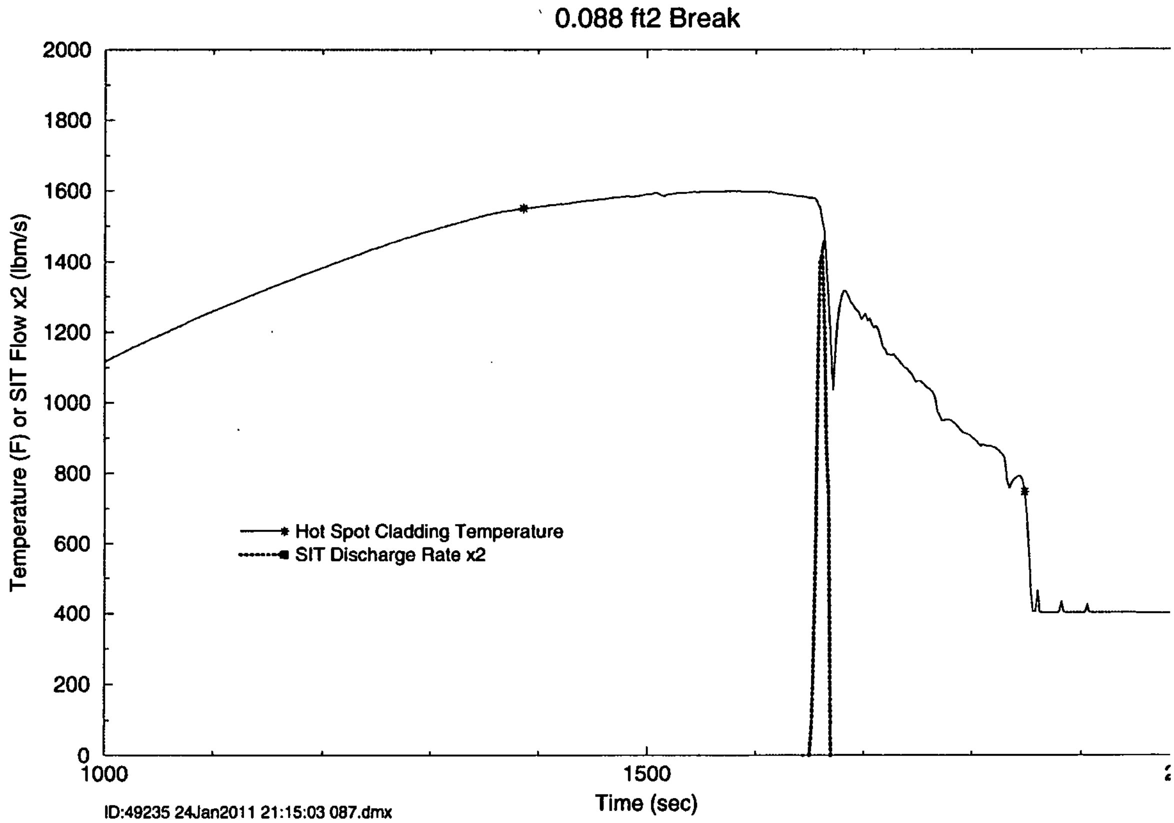


Figure 1-2, Peak Cladding Temperature for a 0.088 ft^2 Break

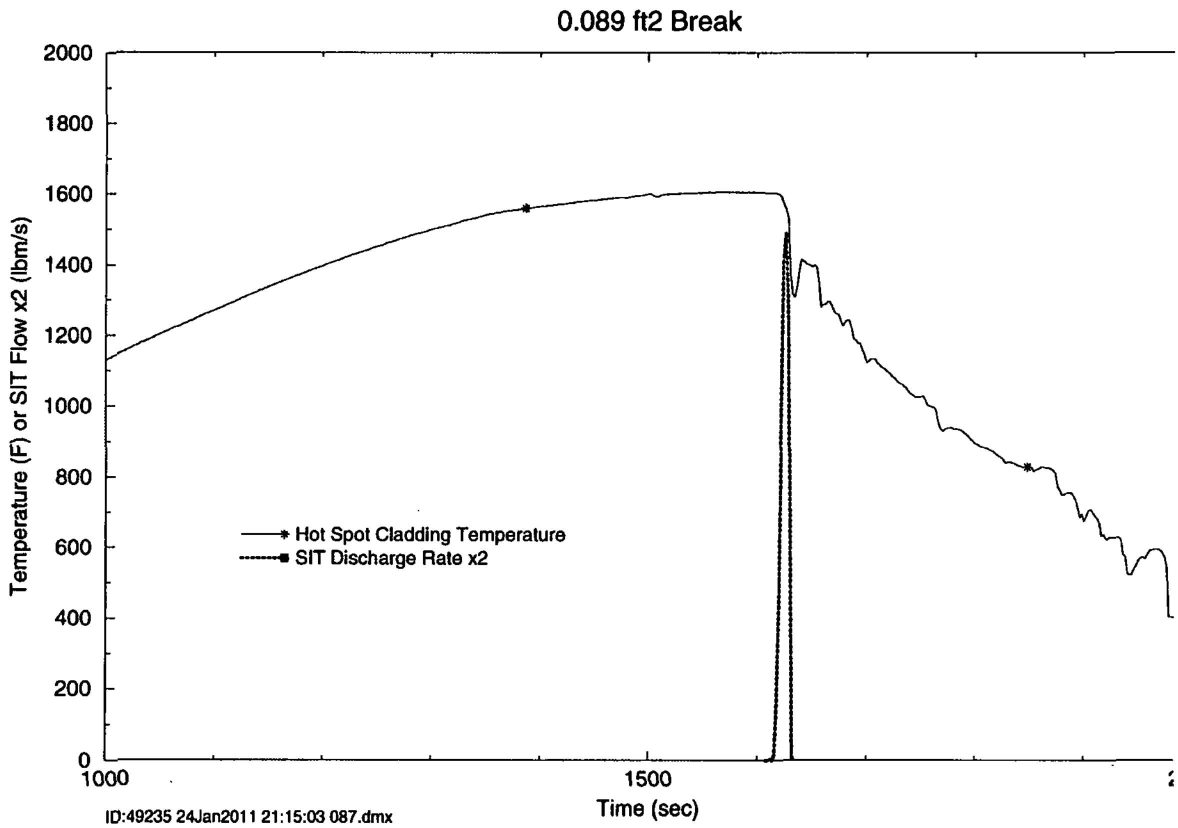


Figure 1-3, Peak Cladding Temperature for a 0.089 ft^2 Break

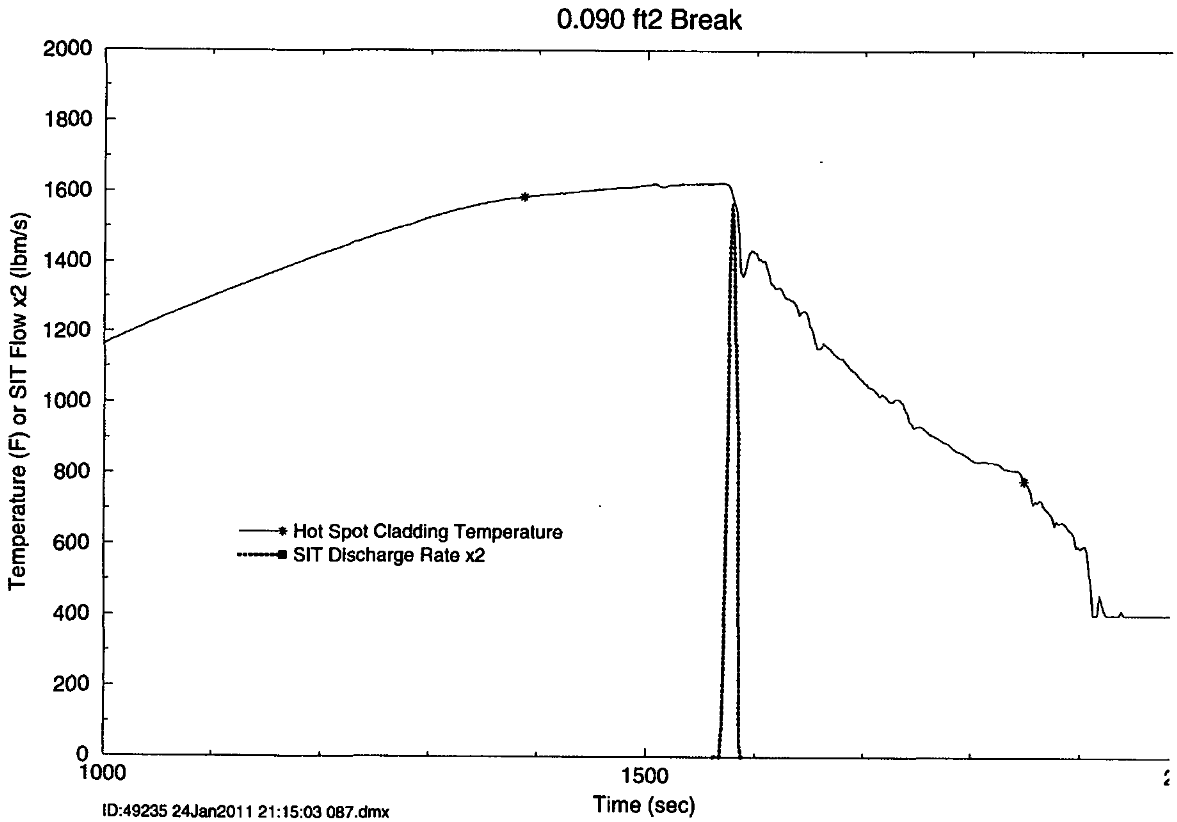
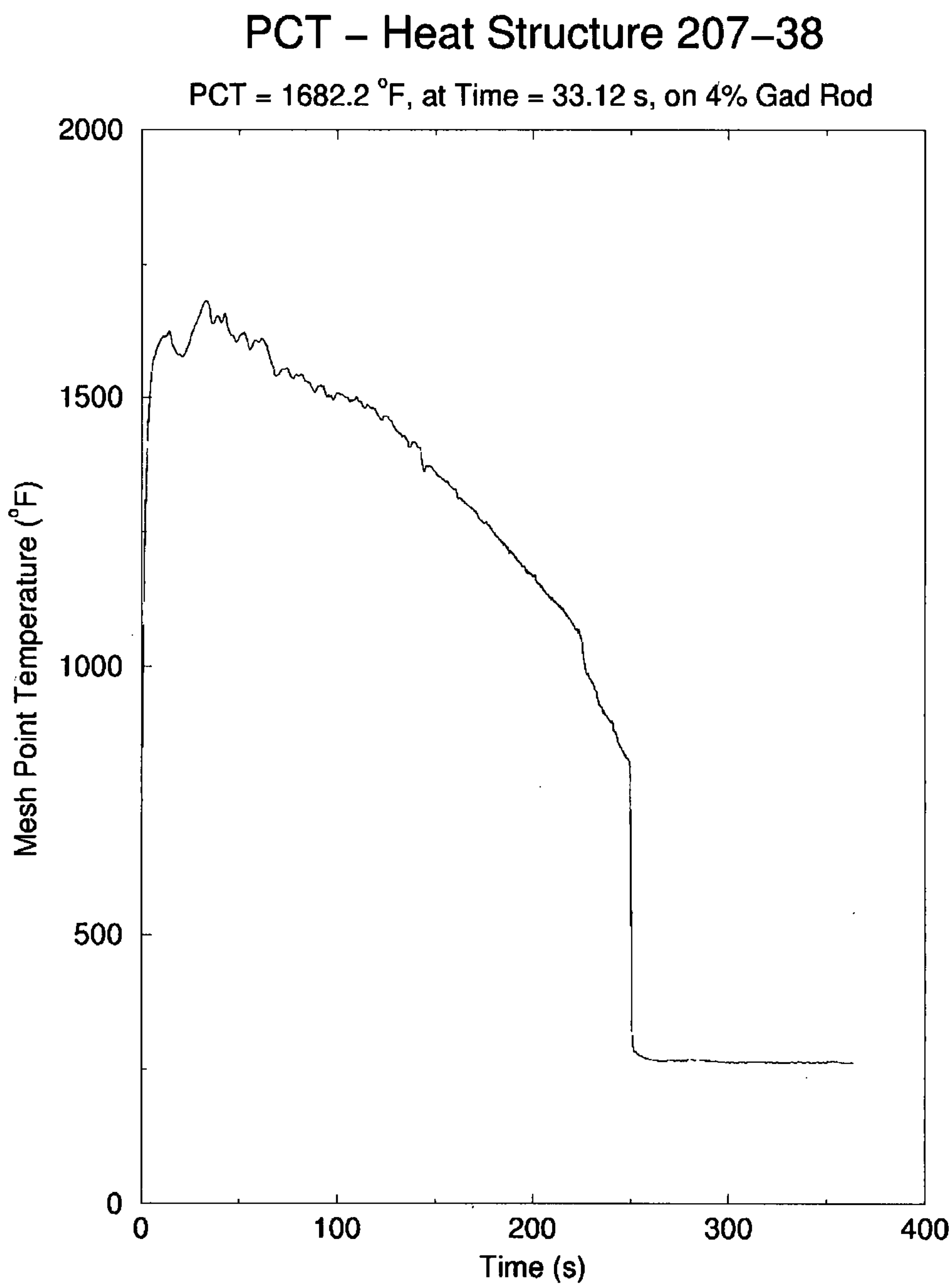


Figure 1-4, Peak Cladding Temperature for a 0.090 ft^2 Break

Supplemental Information 2

The figures below are provided in connection with discussions regarding the Response to Question 33 provided in Reference (1).



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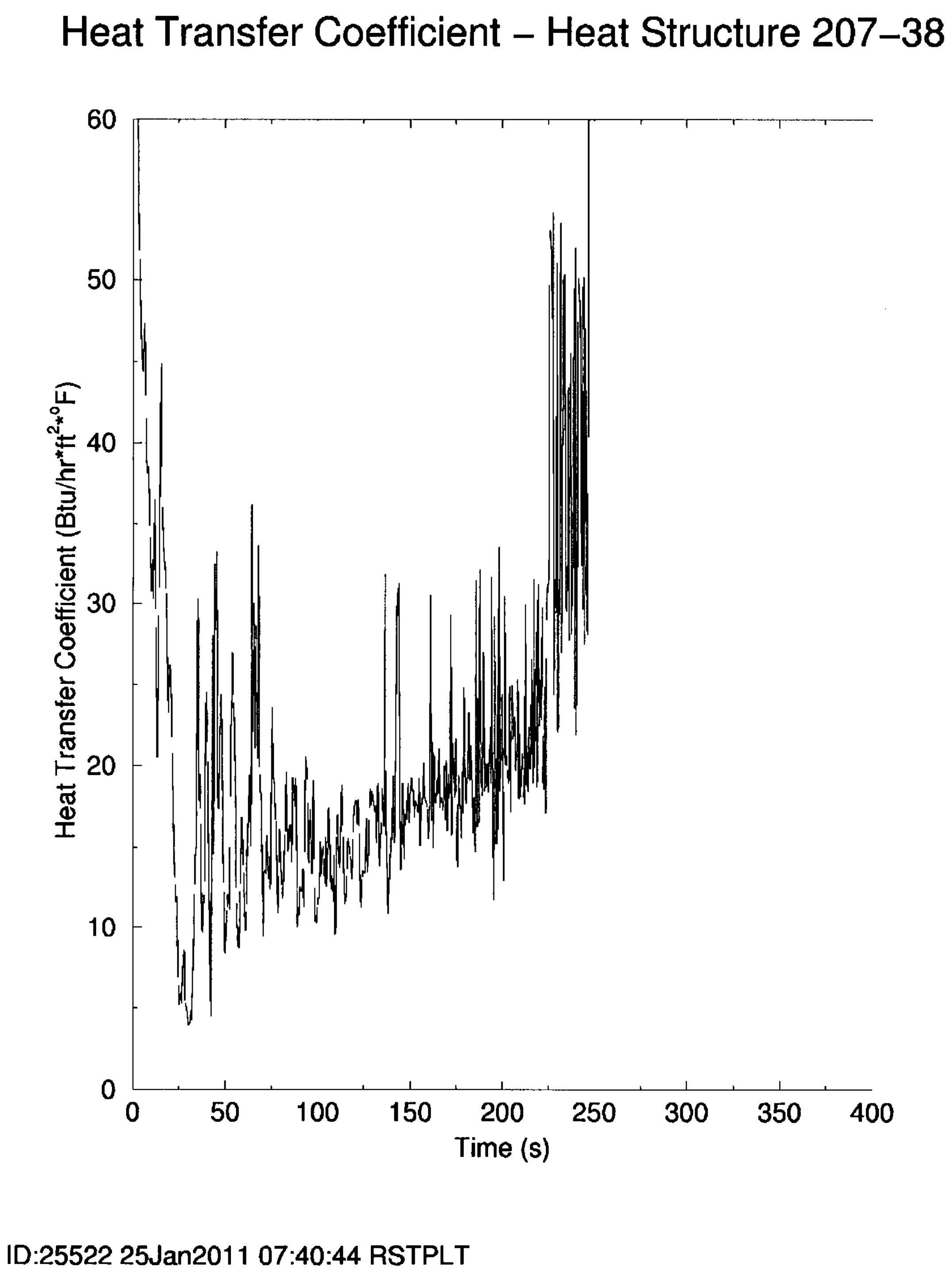
[ ]


Heat Transfer Coefficient – Heat Structure 207–38
Heat Transfer Coefficient (Btu/hr*ft^2*°F)
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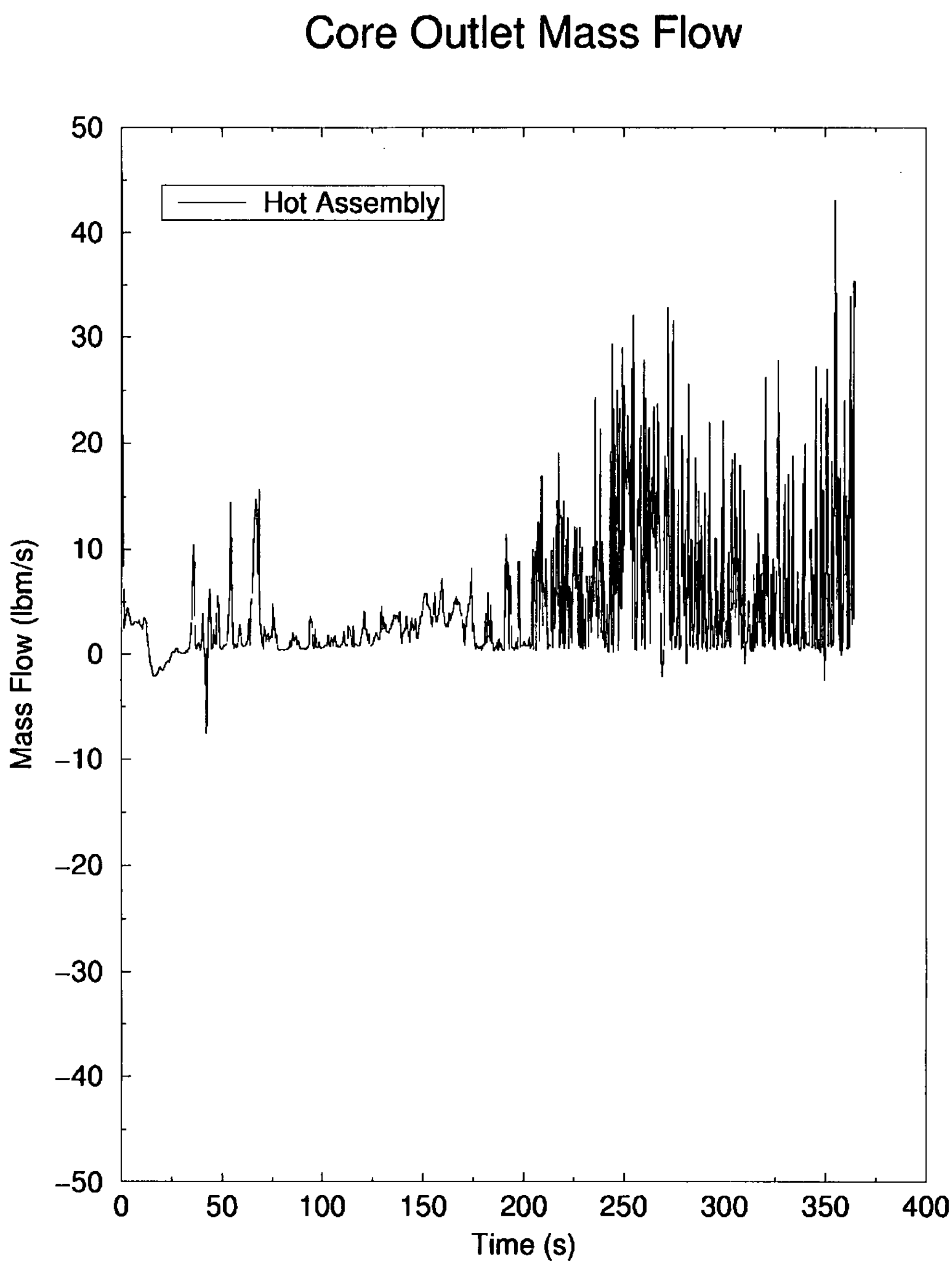
10
20
30
40
50
60
0
50
100
150
200
250
300
350
400
Time (s)
ID:25522 25Jan2011 07:40:44 RSTPLT



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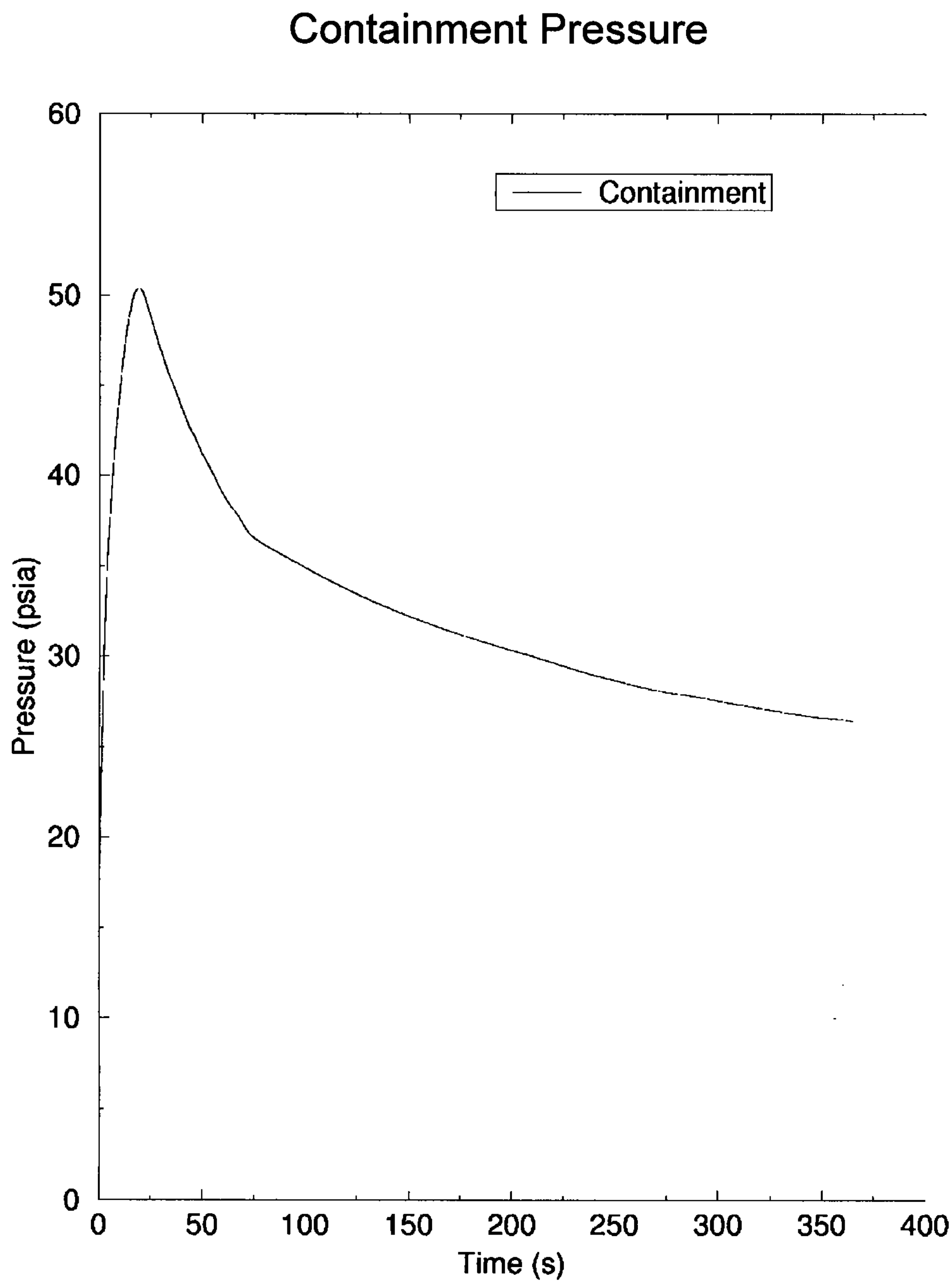

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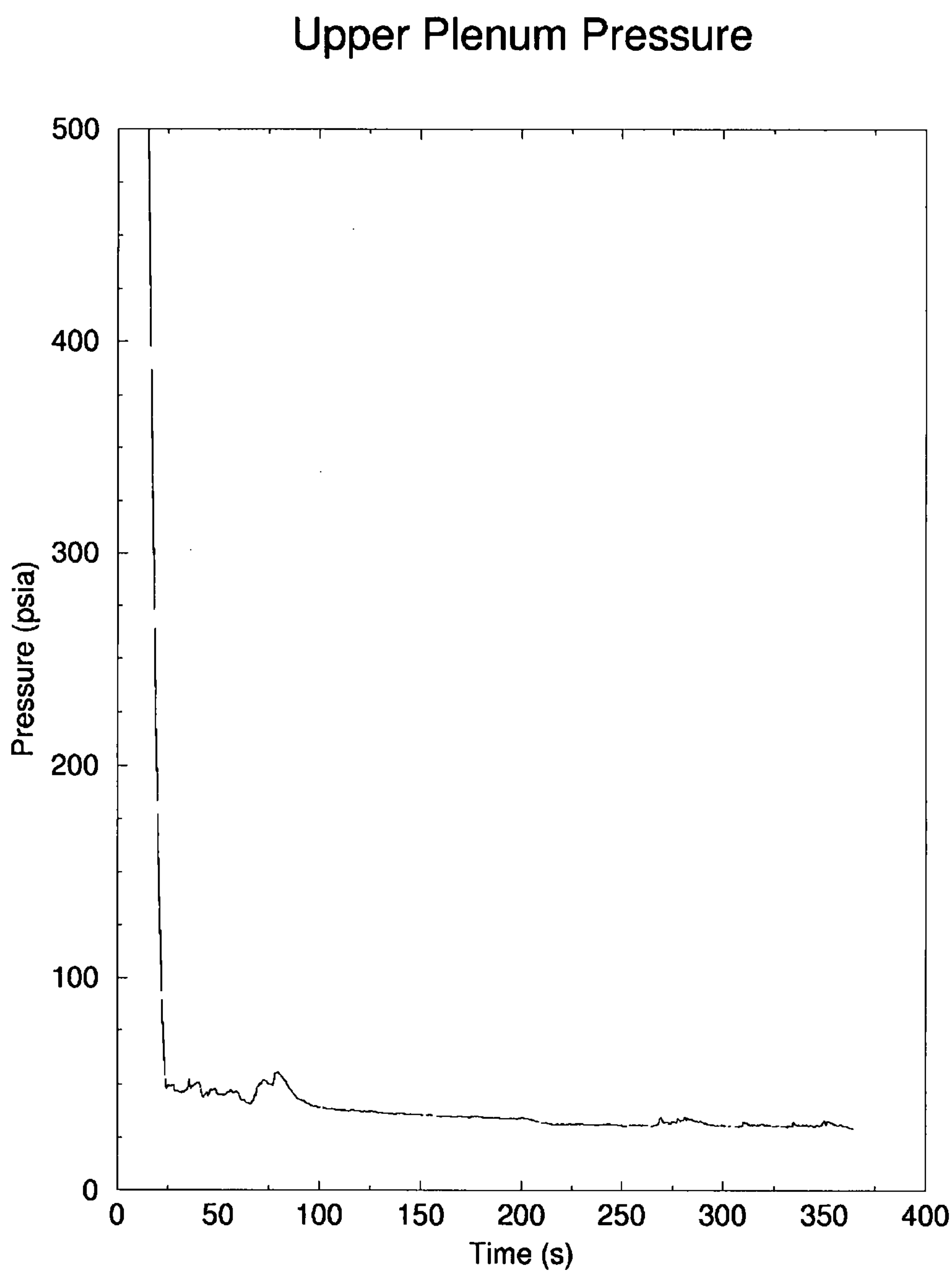

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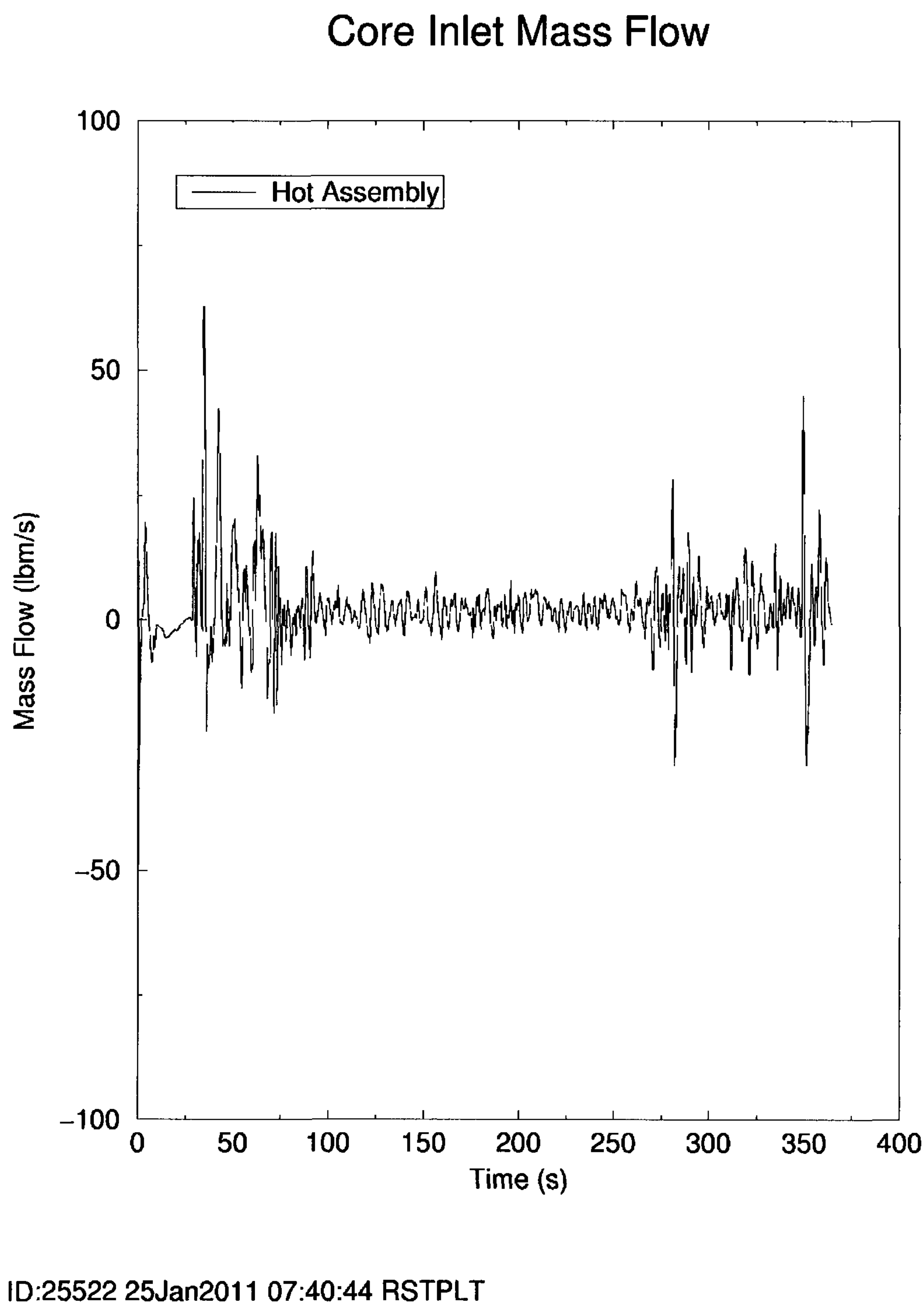
Core Inlet Mass Flow
Hot Assembly
Mass Flow (lbm/s)
Time (s)
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50
0
−50
−100
0
50
100
150
200
250
300
350
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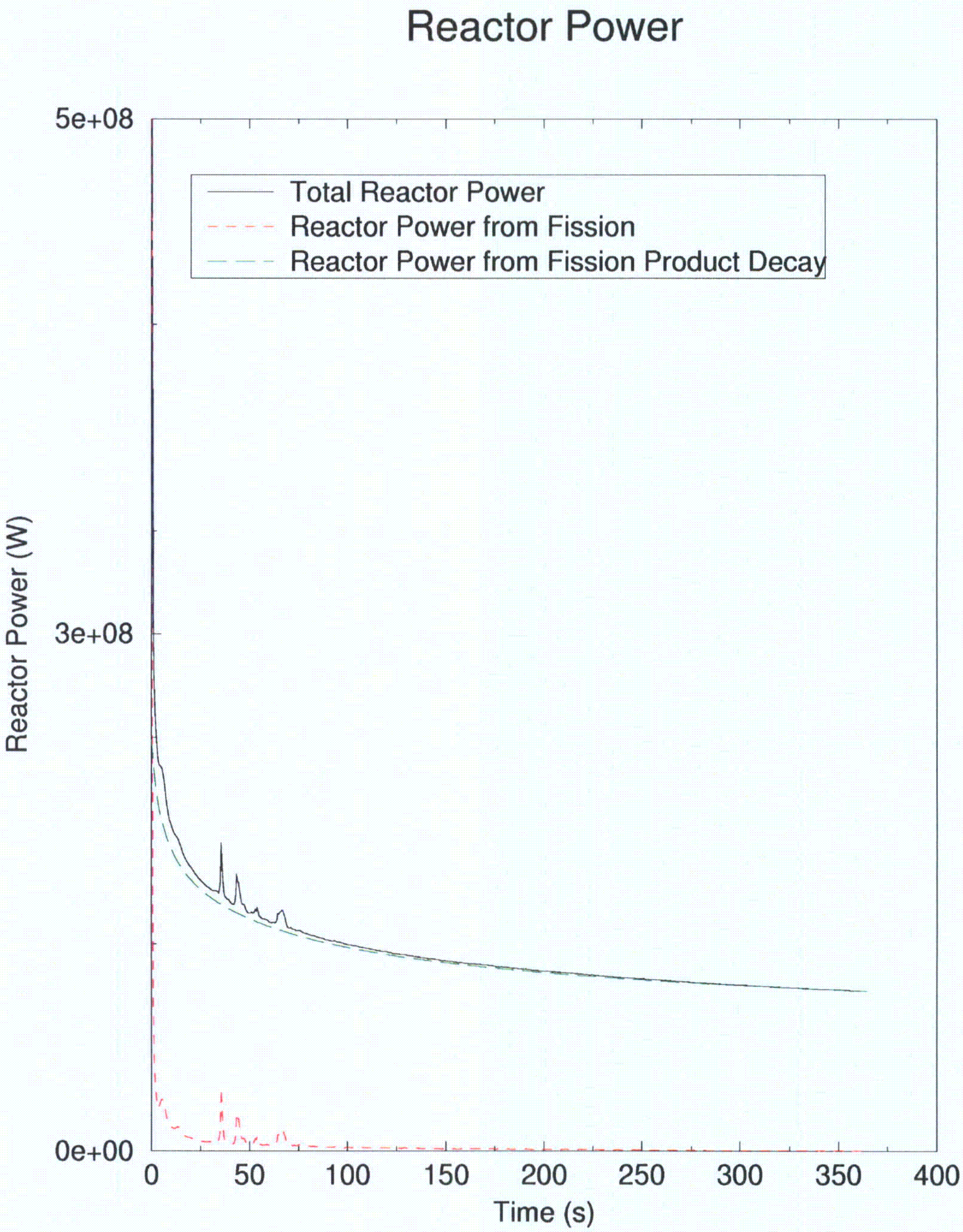

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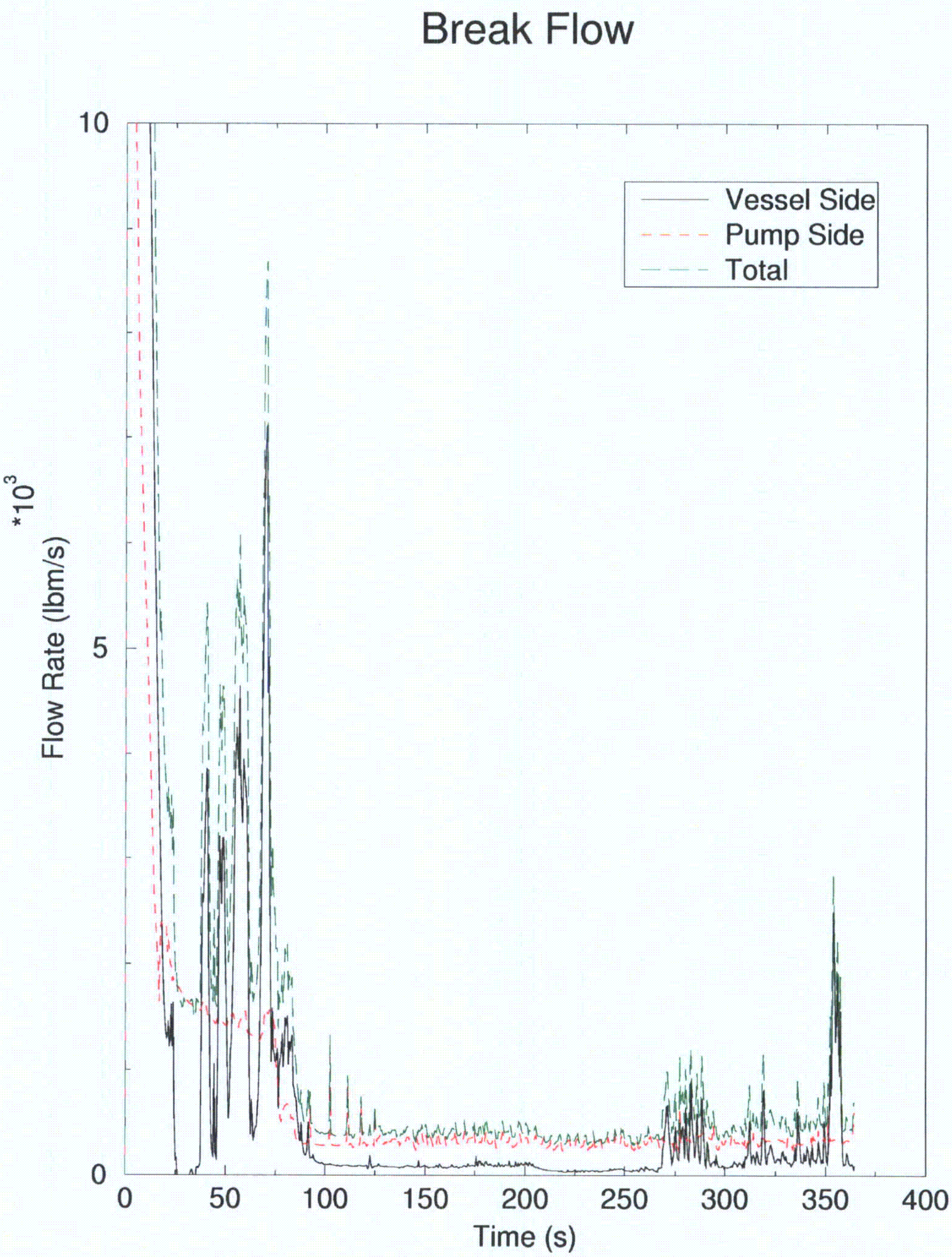

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Supplemental Information 3

In connection with continued discussions of the Response to Question 12 provided in Reference (2), the following information is provided.

The mixture level shown in Figure 3-2 locates an approximate elevation where there is a large void discontinuity. It is not used in the code for any system calculations. The nodal average void fraction along with the non-equilibrium and non-homogeneous fluid conditions are used in the actual clad thermal calculation.

The S-RELAP5 determines where the mixture level is in vertically-oriented volumes. The mixture level within that volume is determined from the void fractions above and below the detected mixture level.

For the period of interest, this translates to [ ]. Note the volumes in this region are [ ] tall. This is a calculated mixture level that is based on [ ]; i.e., even though the lower volumes have begun to accumulate more liquid (void fraction decreasing), it has essentially no effect on the volume in which the calculated mixture level resides.

The mixture level estimated above agrees well with the code-calculated mixture level shown in Figure 3-2. This is not an actual mixture height as one would derive from a code with a true drift flux or level swell model, but a reasonably accurate approximation from which the calculation can be understood. Direct interpretation of the mixture height from the void fraction distribution gives the same approximate result within one core volume.

Figure 3-2 also shows that the core inlet flow slightly exceeded [ ], evidenced by the increase in the core liquid level (decrease in the core entrance voiding shown in Figure 3-1) and also the increase in the downcomer collapsed liquid level. So, there is little or no change in the core mixture level during this time and the level stays relatively constant with the temperature increasing slightly until just before the mixture advances one S-RELAP node, at around [ ].

Figure 3-3 shows that during this period [ ], the total high pressure safety injection flow rate and the break flow rate crossed, with the high pressure safety injection exceeding break flow after about [ ]. This is consistent with the core and downcomer level response shown in Figure 3-2.

The adjustment of mixture level established sufficient cooling to slow the hot spot temperature rise as shown in the cladding thermal response shown in Figure 5-13 in the calculation file. The SIT injection then increased the supply of cooling water, lowered the system pressure, and established a clear path to the end of the temperature excursion.

[ ]

Figure 3-1, Void Fractions - Two-Phase Mixture Level

[ ]

Figure 3-2, Two-Phase Mixture and Collapsed Liquid Levels in the Core

Figure 3-3, Break, Total High Pressure Safety Injection, and Core Inlet Flow Rates

References

1. Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated December 30, 2010, Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel
2. Letter from Mr. T. E. Trepanier (CCNPP) to Document Control Desk (NRC), dated January 14, 2011, Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel