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Your ref: Docket No. 52-006
Our ref: DCP/NRC1667

January 9, 2004

SUBJECT: Transmittal of Responses to AP1000 DSER Open Items

This letter transmits the Westinghouse responses to Open Items in the AP1000 Design Safety Evaluation Report (DSER). A list of the DSER Open Item responses transmitted with this letter is Attachment 1. The proprietary responses are transmitted as Attachment 2. The non-proprietary responses are provided as Attachment 3 to this letter.

The Westinghouse Electric Company Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit are also enclosed with this submittal letter as Enclosure 1. Attachment 2 contains Westinghouse proprietary information consisting of trade secrets, commercial information or financial information which we consider privileged or confidential pursuant to 10 CFR 2.790. Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosures.

This material is for your internal use only and may be used for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Commission, the Office of Nuclear Reactor Regulation, the Office of Nuclear Regulatory Research and the necessary subcontractors that have signed a proprietary non-disclosure agreement with Westinghouse without the express written approval of Westinghouse.

D063

January 9, 2004

Correspondence with respect to the application for withholding should reference AW-04-1766, and should be addressed to James A. Gresham, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania, 15230-0355.

Please contact me at 412-374-4728 if you have any questions concerning this submittal.

Very truly yours,

R. P. Vijuk

R. P. Vijuk, Manager
Passive Plant Engineering
AP600 & AP1000 Projects

/Enclosure

1. Westinghouse Electric Company Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit AW-04-1766.

/Attachments

1. List of the AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses transmitted with letter DCP/NRC1667
2. Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated January 9, 2004
3. Non-Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated January 9, 2004

DCP/NRC1667
Docket No. 52-006

January 9, 2004

Enclosure 1

Westinghouse Electric Company
Application for Withholding and Affidavit

Westinghouse

Westinghouse Electric Company
Nuclear Power Plants
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

January 9, 2004

AW-04-1766

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Mr. John Segala

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: Transmittal of Westinghouse Proprietary Class 2 Documents Related to AP1000 Design Certification Review Draft Safety Evaluation Report (DSER) Open Item Response

Dear Mr. Segala:

The application for withholding is submitted by Westinghouse Electric Company, LLC ("Westinghouse") pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject documents. In conformance with 10 CFR Section 2.790, Affidavit AW-04-1766 accompanies this application for withholding 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 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-04-1766 and should be addressed to the undersigned.

Very truly yours,

R. P. Vijuk, Manager
Passive Plant Engineering
AP600 & AP1000 Projects

/Enclosures

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared James W. Winters, 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 W. Winters, Manager
Passive Plant Projects & Development
Nuclear Power Plants Business Unit

Sworn to and subscribed
before me this 9th day
of January, 2004

Lorraine M. Piplica
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Passive Plant Projects & Development, of the 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 rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company, LLC.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company, LLC 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 2.790 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 which 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 2.790, 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 Attachment 2 as Proprietary Class 2 in the Westinghouse Electric Co., LLC document: (1) "AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Response."

This information is being transmitted by Westinghouse's letter and Application for Withholding Proprietary Information from Public Disclosure, being transmitted by Westinghouse Electric Company letter AW-04-1766 to the Document Control Desk, Attention: John Segala, CIPM/NRLPO, MS O-4D9A.

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

(a) Provide documentation supporting determination of APP-GW-GL-700, "AP1000 Design Control Document," analysis on a plant specific basis

(b) Provide the applicable engineering evaluation which establishes the Tier 2 requirements as identified in APP-GW-GL-700.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for Licensing Documentation.

(b) Westinghouse can sell support and defense of AP1000 Design Certification.

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 methodologies 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 for performing and analyzing tests.

Further the deponent sayeth not.

January 9, 2004

Copyright Notice

The documents transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies for the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.790 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection not withstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond these necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

January 9, 2004

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 2.790 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 2.790(b)(1).

Attachment 1

List of

Proprietary and Non-Proprietary Responses

Table 1 **"List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1667"**	
4.5.1-1 Revision 1 5.3.3-1 Revision 1 15.2.7-1 Item 7 Revision 3 15.2.7-1 Item 12 Revision 1 *Proprietary	*21.5-2P Item APEX Scaling 21.5-2 Item APEX Scaling *CD-ROM for ASCII Data for DSER OI 21.5-3 Item 9

DCP/NRC1667
Docket No. 52-006

January 9, 2004

Attachment 3

AP1000 Design Certification Review
Draft Safety Evaluation Report Open Item Non-Proprietary Responses

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 4.5.1-1 Response Revision 1

Original RAI Number(s): 252.001

Summary of Issue:

The recent experience with VHP nozzle cracking has identified the need for baseline inspection data to determine if an indication is service-induced cracking, or an artifact from fabrication. The staff requested information on what preservice examinations will be performed on the VHP nozzles. In a letter dated April 7, 2003, the applicant responded that preservice examinations for the closure head will include a baseline top-of-the head visual examination, ultrasonic examinations of the inside diameter surface of each vessel head penetration, eddy current examination of the surface of the head penetration welds and the inside diameter surface of the penetrations, and post-hydro liquid penetrant examinations of accessible surfaces that have undergone preservice inspection eddy current examinations. Any indications exceeding the ASME Code Section III requirements would be removed. The information in the RAI response has been provided in DCD Tier 2 Section 5.3.4.7. The information on preservice examinations also needs to be addressed by a COL applicant, and should be reflected in DCD Tier 2 Section 5.3.6, "Combined License Information." This is identified as Open Item 4.5.1-1 and COL Action Item 4.5.1-1.

Westinghouse Response:

The Combined License applicant commitment to specific preservice examinations of the reactor vessel closure head will be added to the existing commitment in DCD section 5.2.6.2 for the Combined License applicant to provide a plant specific preservice inspection program.

Design Control Document (DCD) Revision:

From DCD Revision 5, page 5.2-32:

5.2.6 Combined License Information Items

5.2.6.1 ASME Code and Addenda

The Combined License applicant will address in its application the portions of later ASME Code editions and addenda to be used to construct components that will require NRC staff review and approval. The Combined License applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code edition and addenda added as part of the Combined License application. The Combined License applicant will address the addition of ASME code cases approved subsequent to design certification.

5.2.6.2 Plant Specific Inspection Program

The Combined License applicant will provide a plant-specific preservice inspection and inservice inspection program. The program will address reference to the edition and addenda of the ASME Code Section XI used for selecting components subject to examination, a description of the components exempt from examination by the applicable code, and drawings or other descriptive information used for the examination.

The preservice inspection program will include examinations of the reactor vessel closure head equivalent to those outlined in DCD section 5.3.4.7.

PRA Revision:

None

NRC Comment from 12/17/03 Status meeting:

Issue: It is not clear if the entire volume of the nozzle is subject to preservice examination. Likewise, it is not clear how much of the outside and inside diameter surfaces will be subject to eddy current examination. Please clarify.

Westinghouse Response (Revision 1):

USNRC Order EA-03-009 currently specifies:

a) "Bare metal visual examination of 100% of the RPV head surface (including 360° around each RPV head penetration nozzle), AND

b) Either:

i. Ultrasonic testing of each RPV head penetration nozzle (i.e., nozzle base material) from two (2) inches above the J-groove weld to the bottom of the nozzle and an assessment to determine if leakage has occurred in the interference fit zone, OR

ii. Eddy current or dye penetrant testing of the wetted surface of each J-groove weld and RPV head penetration nozzle base material to at least two (2) inches above the J-groove weld.

The preservice examination program proposed for the AP1000 reactor vessel head penetrations will use a combination of examination techniques to provide coverage which meets or exceeds the requirements of the Order.

For the control rod drive mechanisms (CRDM), preservice ultrasonic examination coverage will include the entire volume of each penetration nozzle from an elevation at least two inches above the J-groove attachment weld to an elevation just above the thread relief on the ID surface of the nozzle. This scan plan will provide volumetric coverage of each penetration nozzle to at least one inch below the J-groove attachment weld.

Preservice eddy current coverage on the OD surfaces of the CRDMs will include the entire J-groove weld and the entire penetration nozzle OD surface from the elevation of the J-groove weld to the bottom of the tube.

For the incore instrumentation (ICI) penetration nozzles, preservice ultrasonic examination coverage will include the entire volume of each penetration nozzle from an elevation at least two inches above the J-groove attachment weld to an elevation at least one inch below the J-groove attachment weld.

Preservice eddy current coverage on the OD surfaces of the incore instrumentation (ICI) penetration nozzles will include the entire J-groove weld and the entire penetration nozzle OD surface from the elevation of the J-groove weld to an elevation two inches from the bottom of the tube.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 5.3.3-1 Response Revision 3

Original RAI Number(s): 251.018

Summary of Original Issue:

The staff requested, in RAI 251.018, that the applicant demonstrate that the P-T limits are in accordance with Appendix G to 10 CFR Part 50. The applicant responded, that the AP1000 heatup and cooldown operating curves were generated using the most limiting adjusted reference temperature values and the NRC-approved methodology as documented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," with staff approved exceptions.

One exception is that instead of using best estimate fluence values, the applicant is using fluence values that are calculated fluence values. The staff finds this acceptable because this is in compliance with RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The other exception is that the KIc critical stress intensities are used in place of the KIa critical stress intensities. This methodology is taken from staff approved ASME Code Case N-641. The staff found the applicant's responses acceptable because the AP1000 P-T limit curves were developed in accordance with 10 CFR Part 50, Appendix G, with the exception that the flange requirement is in accordance with WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." Currently, the staff has not approved WCAP 15315. Any changes to the RV closure head requirements would be incorporated into Appendix G of 10 CFR Part 50. If a relaxation to 10 CFR Part 50, Appendix G is approved, this will allow the operating window to be wider. Since applicants using AP1000 are required to meet the requirements of 10 CFR Part 50, Appendix G, applicants using AP1000 must meet the closure head requirements of Appendix G of 10 CFR Part 50. However, the AP1000 DCD does not provide limitations (values of RTNDT) for the closure flange region of the RV and head. The AP1000 design must include these limitations in order to satisfy Appendix G of 10 CFR Part 50. The applicant should provide these limitations that are consistent with the present TSs and 10 CFR Part 50, Appendix G, or provide closure flange limitations with new TSs that are consistent with 10 CFR Part 50, Appendix G. This is Open Item 5.3.3-1.

NRC Comment From December 17, 2003 NRC/Westinghouse Meeting:

In Tier 1 ITAAC 2.3.6, P/T should be spelled out.

Westinghouse Response (Revision 3):

Westinghouse will modify DCD Tier 1 Table 2.3.6-4 to clearly define the meaning of "P/T", as shown below.

The Westinghouse Revision 3 response addresses only the NRC comment from the December 17, 2003 meeting. The Westinghouse Revision 2 response to this Open Item (as transmitted by Westinghouse letter DCP/NRC1601, July 3, 2003) remains valid except for the change to DCD Tier 1 Table 2.3.6-4 shown below.

Design Control Document (DCD) Revision:

From DCD Revision 8, Tier 1, Section 2.3.6, Table 2.3.6-4, page 2.3.6-12:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.a) The RNS provides LTOP for the RCS during shutdown operations.	i) Inspections will be conducted on the low temperature overpressure protection relief valve to confirm that the capacity of the vendor code plate rating is greater than or equal to system relief requirements.	i) The rated capacity recorded on the valve vendor code plate is not less than the flow required to provide low-temperature overpressure protection for the RCS, as determined by the LTOPS evaluation based on the pressure-temperature curves developed for the as-procured reactor vessel material.
	ii) Testing and analysis in accordance with the ASME Code Section III will be performed to determine set pressure.	ii) A report exists and concludes that the relief valve opens at a pressure not greater than the set pressure required to provide low-temperature overpressure protection for the RCS, as determined by the LTOPS evaluation based on the pressure-temperature curves developed for the as-procured reactor vessel material.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 15.2.7-1 Item 7 Response Revision 3

Original RAI Number(s): None

Summary of Issue:

The revised DCD Section 15.6.5.4C (DSER OI 15.2.7-1P Page 14) states that the LTC phase analysis uses the NOTRUMP DEDVI case at 25 psia containment pressure reported in Section 15.6.5.4B as initial conditions, and the WGOTHIC analysis of this event as boundary conditions.

Please describe the model used to develop the containment backpressure and demonstrate that it represents a bounding and conservative estimate of containment pressure following a small break LOCA. Discuss any differences that may exist between this model and that used in the large break LOCA analyses. Please discuss how water spillage from a broken DVI line is mixed with the containment atmosphere and justify that the treatment is consistent with the Westinghouse ECCS evaluation model. Discuss the conservative treatment of non-safety related containment sprays and containment coolers in reducing containment pressure. Please also clarify if the 25 psia initial condition is consistent with the WGOTHIC analysis of the containment pressure as a function of time.

Westinghouse Response (Revision 3):

Attachment B provides Westinghouse response to three items identified in the 12/17/03 Open Item status meeting.

Westinghouse Response (Revision 2):

As a result of the assessment of the Westinghouse AP1000 small-break WGOTHIC containment model by the NRC Containment and Accident Dose Assessment Section (Ref: "An Assessment of the Westinghouse AP1000 Small-Break LOCA WGOTHIC Containment Model for Minimum Containment Pressure and Lon-Term Cooling," USNRC, November 2003), Revision 2 of this response has been prepared. Attachment A summarizes the NRC assessment and provides the Westinghouse response.

Original and Revision 1 Response:

For AP600 and AP1000, two different WGOTHIC models were used to determine the containment backpressure that would exist following a LOCA event. Assumptions were used in these models to conservatively underpredict the pressure. These two models are discussed below.

Large Break LOCA Model for PCT Calculation

For this case, a simplified WGOTHIC model of the containment was developed to determine the containment pressure response during the blowdown portion of a double-ended cold leg break.

This model consists of a single control volume that represents the containment, all the heat sinks inside containment, and a simplified thermal conductor representing the containment shell that is connected from the containment control volume to a control volume that represents the environment. The boundary conditions for this model are specified in Reference 1. The outside temperature of the shell is held at a constant temperature of 0F. The heat transfer coefficient inside containment consists of the Tagami correlation for the blowdown portion of the transient (first 29 seconds), and the Uchida condensation correlation for the time following blowdown. These heat transfer coefficients are applied on all the internal heat sinks as well as the inside of the containment shell. As specified in Reference 1, the Tagami correlation is multiplied by a factor of four, and the Uchida correlation is multiplied by a factor of 1.2.

Small Break LOCA Model to Determine Containment Backpressure

A second WGOTHIC model was used to determine the AP600 and AP1000 containment backpressure after a small break LOCA event. These results were used as the boundary conditions for small break LOCA NOTRUMP analyses and WCOBRA/TRAC long term cooling analyses. The model is the same as the evaluation model used to determine the peak containment pressure for the DCD with assumptions changed to minimize the pressure response. For AP600, this model was used to support the long term cooling analysis, but was not used for the small break LOCA backpressure. For AP1000, this model was used to support the long term cooling analysis as well as the double-ended DVI (DEDVI) break analysis.

This model is described in Reference 2 which was submitted to the NRC and reviewed as part of AP600 Design Certification. As specified in Reference 2, the following changes to the DCD WGOTHIC model were made for this analysis:

a) The DCD model is biased to maximize containment pressure. These assumptions were changed for the backpressure analysis to minimize containment pressure.

- Heat transfer coefficient multipliers which are set to values less than unity for the peak pressure analysis are set to unity for the backpressure analysis
- Heat sinks that are conservatively neglected for the peak pressure analysis are included for the backpressure analysis
- Initial conditions inside containment that are biased to the highest operating pressure and temperature are set to the lowest operating pressure and temperature. Relative humidity is set to 100% to minimize the initial air inventory inside containment. Environmental boundary conditions are biased to maximize heat transfer from the passive containment cooling system and minimize the containment pressure.
- The containment vent system is assumed to be open at the start of the event and closes on an SI signal. This allows an initial decrease in the air inventory which results in a lower containment pressure

b) Mass and energy release rates that are specific for the double-ended DVI break are included in the WGOTHIC model.

- Water spilling from the broken DVI is assumed to enter the PXS compartment containing the break. This water does not interact with the containment atmosphere as it falls from the break.

Non-safety systems such as containment fan coolers and containment sprays are not considered for this analysis. The containment spray system is only used in the event of severe accidents. Its use requires the operator to align the pumps and water sources for operation (requires an operator to open manual valves out in the plant). The chilled water supply to the fan coolers is automatically isolated following an SI signal. The system can be restarted by the operator to assist in long-term recovery following a LOCA, and it is not considered in this shorter-term analysis.

Two sensitivity studies were done to determine the effect of these assumptions.

Cold Water Spill Sensitivity

The DEDVI break consists of two break flow paths; one from the vessel side, and one from the loop side. The vessel side break is a typical high-temperature, high-pressure two-phase blowdown. This two-phase flow is assumed to form droplets that are dispersed into the atmosphere of the break compartment. The recommended drop size is 100 microns. The loop side break flow consists of low-temperature (~80F), high-pressure single phase water. Normally, this water is spilled to the floor of the compartment and is not assumed to interact with the vapor-space region of the compartment. For this sensitivity study, the loop side break flow is assumed to be dispersed into the atmosphere of the break compartment.

Figure 1 shows the containment pressure response for the two cases. By allowing the cold water to interact with the steam and two-phase mixture in the compartment vapor space, the overall pressure is reduced by approximately 2 psi. The pressure remains above 25 psia between the time that the ADS4 flow becomes non-critical and the time of IRWST injection. The interaction between the steam and the water droplets in the compartment causes steam to condense resulting in less steam to pressurize the containment. In addition, the water droplets are heated so that the water accumulating on the compartment floor is saturated.

Heat Transfer Coefficient Sensitivity

The Tagami correlation is not considered appropriate for use in small break LOCA analysis. This correlation was developed to account for significant forced convection heat transfer that takes place during the blowdown period of a large break LOCA. This time period is about 30 seconds. For the DEDVI, the "blowdown" period extends to about 500 seconds during which an equivalent amount of energy is released from the RCS to the containment atmosphere as occurs for the large break LOCA. Since the forced convection in containment depends on a characteristic velocity, the velocities inside containment during the blowdown period can be compared for the two events by comparing the blowdown time. Thus, it is likely that the velocities would be at least a factor of ten lower for the DEDVI than for the DECL during the blowdown, and since the forced convection heat transfer coefficient is roughly proportional to the velocity, use of Tagami during a small break LOCA blowdown would significantly overpredict the forced convection heat transfer. The Uchida correlation is recommended for these analyses.

As a sensitivity, the multiplier on the Uchida correlation was increased to 4.0 during the blowdown period (<500 seconds). Figure 2 shows the containment pressure response with and without this multiplier assuming mixing of the cold water spill as described above. These results show little sensitivity to the increased heat transfer coefficient.

The results of these sensitivity studies show that the containment backpressure boundary condition of 25 psia is valid for use in the DEDVI small break LOCA analysis.

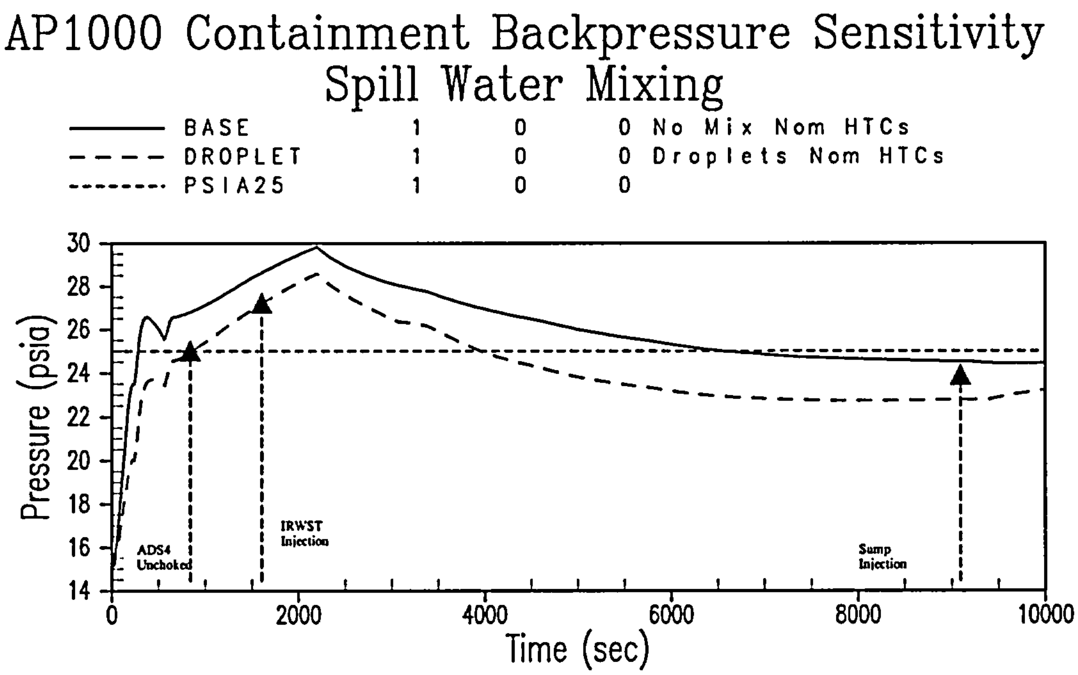


Figure 1: Cold Water Droplet Size Sensitivity

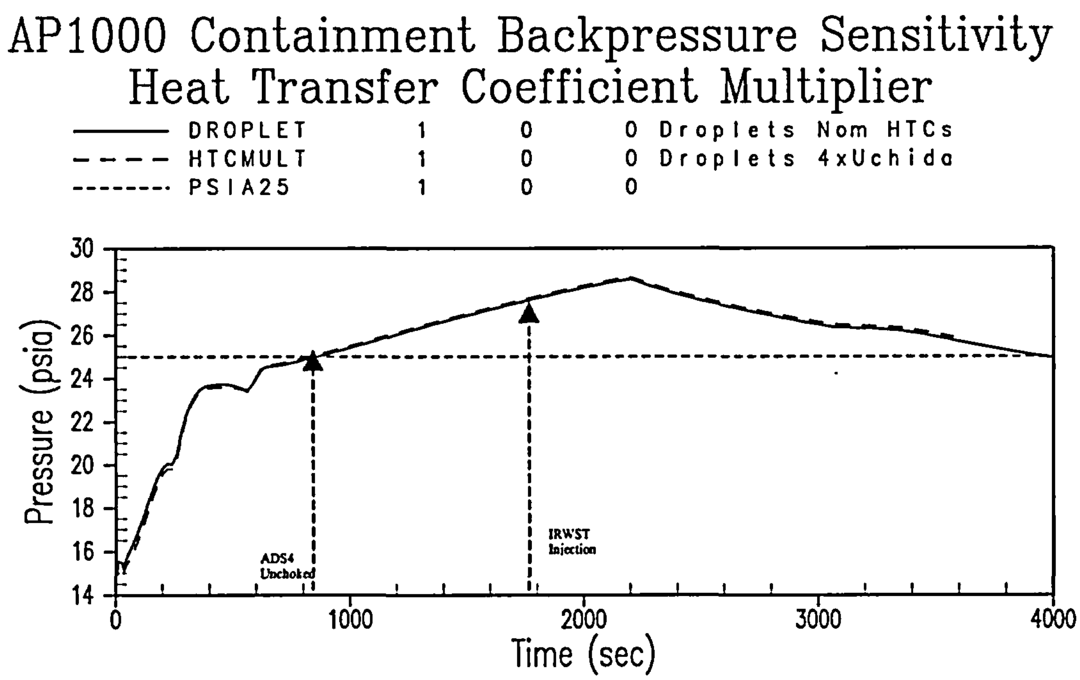


Figure 2: Heat Transfer Multiplier Sensitivity

References

1. WCAP-14171, Rev 2, WCOBRA/TRAC Applicability to AP600 Large Break LOCA March 1998.

2. WCAP-14601, Rev 2, AP600 Accident Analysis Evaluation Models, May 1998.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Attachment A

Summary of NRC Assesment

Bounding Containment Backpressure for AP1000 Long-Term Cooling

The NRC staff asked for a description of the model used to develop the containment backpressure for small-break LOCA and long-term cooling (DSER Open Item 15.2.7-1 Item 7 [Ref 1]). In addition, the staff asked for a discussion of the differences between this model and the model used for minimum pressure following a large-break LOCA. Specifically, the staff asked Westinghouse to discuss how cold water spilled from the broken DVI line interacts with the containment atmosphere, and to discuss the role of non-safety equipment such as fan coolers and containment sprays for these analyses.

The Westinghouse response to this open item is summarized as follows:

1. The WGOTHIC model for small-break LOCA backpressure and long-term cooling is consistent with the methodology in WCAP-14601, "AP600 Accident Analyses – Evaluation Models," Rev 2 (Ref 2).
2. The use of the Tagami correlation is not appropriate for small-break LOCA analyses.
3. Heat sinks that are removed for the peak pressure calculation are included in the small-break LOCA backpressure and long-term cooling calculation
4. Non-safety fan coolers and containment sprays require specific post-accident operator action and would not be available during the time frame prior to sump recirculation switchover.
5. The cold water spill was introduced to the containment atmosphere as a fine mist (100-micron drops) and the resulting containment pressure was reduced by approximately 2 psi.

The Westinghouse response to the Open Item was referred to the Containment and Accident Dose Assessment Section for further review. The NRC staff prepared an assessment of the AP1000 Small-Break LOCA WGOTHIC Containment Model for Minimum Containment Pressure and Long-Term Cooling (Ref. 3). The assessment concluded the following:

1. The AP1000 small-break LOCA and long-term cooling minimum containment pressure methodology is consistent with WCAP-14601, "AP600 Accident Analyses – Evaluation Models," (Ref. 4). Key assumptions for this analysis are summarized below:
    - the containment volume is 1.05 times the best estimate value

- the initial containment temperature is 120F
- the passive heat sink areas are 1.05 times the best estimate values
- the PCS water flow is set to the maximum value and water coverage is set to the maximum value
- no penalties are applied to the PCS heat and mass transfer correlations

2. The AP1000 small-break LOCA and long-term cooling minimum containment pressure methodology differs from the methodology used to determine the minimum pressure for large-break LOCA as described in the AP600 and AP1000 Tier 2 DCD Section 6.2.1.5 (Ref. 5). These differences are summarized below:

   - the containment volume is 1.1 times the reference value
   - the passive heat sink surface areas are 2.1 times their reference values
   - the material properties are biased high
   - the air annulus and containment shell temperatures are assumed to be held at a constant 0F
   - the containment purge is operated at time zero and closes 12 seconds after the pressure setpoint of 8 psig is reached
   - the initial containment pressure and temperature are set to their low values (14.7 psia, 90F) consistent with SRP 6.2.1.5
   - the containment relative humidity is set to 99%
   - the Tagami correlation with a multiplier of 4.0 is used for the blowdown

3. The staff concluded that use of the Tagami correlation was not justified for small-break LOCA. It was recommended that the Uchida correlation with a 1.2 multiplier be used.

4. It is the staff's recommendation that the WGOTHIC small-break LOCA and long-term cooling model be modified to include the most conservative assumptions from the WGOTHIC minimum pressure for large-break LOCA model. Specifically,

   a. The containment net volume should be increased by a factor of 1.1
   b. The containment shell and PCS heat structure area should be increased by a factor of 1.1
   c. The remaining head structure areas should be increased by a factor of 2.1 or a lower value if justified based on an accounting of expected structures in the final as-build plant.
   d. The Uchida correlation with a multiplier of 1.2 should be used for passive heat structures (non PCS structures throughout the accident.
   e. The PCS heat and mass transfer correlation multipliers should be appropriately biased to account for the uncertainty in the experimental data base, and forced convection should be included on the PCS inner surface
   f. Head transfer in dead-ended compartments below the operating deck should not be turned off at the end of blowdown.

g. The air-gap between the steel and the concrete should be reduced from the 20-mill thickness used in the maximum pressure calculation. A zero thickness air-gap would be conservative.
h. The material properties should be biased high for conservatism.
i. Heat transfer credit for the PCS should start earlier than is currently assumed for the maximum pressure calculation.
j. Westinghouse should maintain its treatment of ECCS spillage as implemented in 1979.
k. The containment purge system should be assumed to be operating and isolate on high pressure signal
l. The initial and boundary conditions for the PCS water and environment should be provided with their justification for staff review.

Westinghouse response:

The NRC recommendations listed above were incorporated into the WGOTHIC small-break LOCA and long-term cooling model, and the pressure response to a double-ended DVI break LOCA is shown in Figure A-1 as the NRC case. Also shown as the base case is the pressure using the assumptions from the approved methodology in Reference 2.

The following are Westinghouse comments regarding the recommendations of the staff assessment and discussion of the parameters used in the WGOTHIC analyses for Figure A-1:

a. The main parameters of the volume enclosed by the containment shell are established by an ITAAC that requires the dimension of the shell inside diameter and height above the operating deck. Major structures and components inside containment are also identified in ITAAC. Uncertainties associated with the volume of the structures and equipment inside containment will result in a small uncertainty in the net free volume. The base case uses a multiplier of 1.05 to conservatively account for this uncertainty. The NRC case uses a multiplier of 1.1.

b. The containment shell dimensions (shell inside diameter and height above the operating deck) are established by ITAAC. Also the water film on the external surface of the containment shell is assumed to distribute evenly around the entire circumference of the containment shell to maximize the effectiveness of the PCS heat transfer area. The base case uses a factor of 1.05 on PCS heat transfer area to conservatively account for any residual uncertainty in PCS heat transfer area. The NRC case uses a factor of 1.1 for this uncertainty.

c. The methodology for the WGOTHIC small-break LOCA and long-term cooling containment backpressure was approved for AP600 in Reference 2. The WGOTHIC model for minimum backpressure for large-break LOCA was completed before the passive heat sink information was finalized. Consequently, the passive heat sink area multiplier in the approved methodology, 1.05, was doubled to account for this uncertainty to 2.1. Since the results were within acceptable margins, there was no need to repeat the analysis as the design matured and the passive heat sinks were better defined. This large overconservatism was noted by the staff during review of AP600. The base case in Figure A-1 uses a 1.05 multiplier on heat sinks. The NRC case uses a 2.1 multiplier on heat sinks.

d. As was noted in Reference 3, the use of the Tagami correlation is not appropriate for small-break LOCA. For the base case analysis the Uchida correlation is used with a 1.2 multiplier during blowdown and 1.0 multiplier after blowdown. For the NRC case the Uchida correlation is used with a 1.2 multiplier throughout the accident simulation.

e. The base case uses multipliers of 1.0 on the PCS heat and mass transfer correlations. For the NRC case these multipliers were 1.19 and 1.37. Since WGOTHIC is being used in the lumped-parameter mode, the gas velocity is not known in the cells adjacent to the shell. Considering that velocities will be small for small-break LOCA events, forced convection is not modeled for either case.

f. For both cases heat transfer in the dead-ended compartments was not turned off at the end of blowdown as recommended.

g. The base case includes the air gap thickness. The air-gap thickness was eliminated for the NRC case.

h. The material properties for steel, concrete and air were set to nominal values in the base case. For the NRC case, the properties were biased to maximize heat absorption.

i. The PCS startup time was accounted for in the base case. The PCS was started at full flow at the start of the transient for the NRC case.

j. The base case treats ECCS spillage as liquid flowing to the containment sump without interaction with containment atmosphere. Consistent with the Revision 1 of this Open Item response, for the NRC case the ECCS spillage is treated as a mist with a droplet size of 100 microns to maximize the heat transfer with the containment atmosphere.

k. For both cases the containment purge system is assumed to open and isolates on a high containment pressure signal.

l. For both cases the initial temperature (120F) and the initial humidity (100%) inside containment are biased high to minimize the amount of non-condensable gas at the start of the transient. For both cases the initial temperature and PCS temperature outside containment are biased low (40F) to maximize heat removal from the containment shell.

The NOTRUMP small-break LOCA response, and the WCOBRA/TRAC long term cooling response will be re-analyzed using the NRC case for the AP1000 DEDVI break. The results will be submitted in another revision to this response.

References

1. DSER Open Item 15.2.7-1 Item 7 Revision 1
2. WCAP-14601, "AP600 Accident Analyses – Evaluation Models," Rev 2
3. Assessment of the AP1000 Small-Break LOCA WGOTHIC Containment Model for Minimum Containment Pressure and Long-Term Cooling, E. Throm, USNRC, December 2003.
4. WCAP-14601, "AP600 Accident Analyses – Evaluation Models"

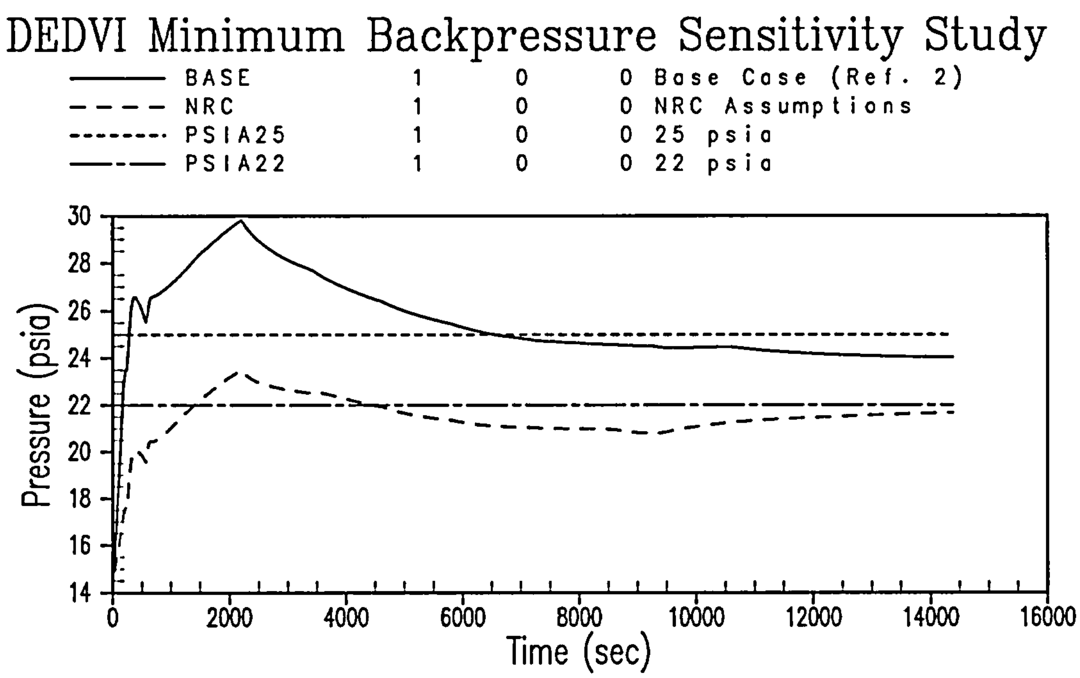


Figure A-1: AP1000 Containment Backpressure Sensitivity

Attachment B

NRC Comments from 12/17/03 Open Item status meeting:

a) Provide complete graphs of minimum containment pressure, minimum containment water level and minimum water temperature as a function of time for 30 days.

The staff calculations show that the ability to cool the core is highly dependant on containment pressure, containment water level, and sump temperature. Westinghouse has provided us with containment pressure only for the first 10000 seconds and then at 14 days and 28.5 days, and the containment water level only at 2.6 hours, 14 days and 28.5 days.

b) Address inadvertent containment spray either from equipment failure or operator error during the 30 day period. Of particular concern is a sufficiently reduced containment pressure that increases the ADS-4 resistance and depresses the two-phase level into the vessel. If the vessel two-phase level is depressed below the top of the hot legs, liquid flow out the ADS-4 valves will decrease causing the boric acid to begin accumulating in the vessel.

c) When the containment pressure issue is settled, Westinghouse needs to redo long term cooling analyzes including those for the PRA using acceptable models.

Westinghouse Response to NRC Comments:

a) An additional bounding WGOTHIC analysis has been performed with NRC assumptions (see Attachment A of this response) and assuming that plant operators re-start the non-safety related containment fan coolers at 10 minutes. The WGOTHIC analysis was extended to 14 and 30 day conditions. The table below provides the resulting containment pressure, sump and PXS compartment water temperatures, and sump level out to thirty days.

Time (days)	Pressure (psia)	Sump Temperature (F)	PXS Compartment Temperature (F)	Sump Level (ft) Grade = 100 ft
0	14.7	N/A	N/A	N/A
0.025 (peak pressure)	22.0	N/A	N/A	N/A
0.104 (beginning of recirculation)	19.0	192	150	108.1
0.5*	17.5	192	158	107.9
1.0*	17.0	193	166	107.7
5.0*	16.1	195	176	105.5
7.0*	16.0	196	178	104.7
14.0	15.7	196	182	103.5
30.0	15.4	196	187	103.5
*Values at these times are estimated by hand calculations.				

b) The containment spray capability in AP1000 is non-safety related and would not be used following an accident unless there were clear indications that a severe accident had occurred, at which point core cooling has already been lost.

The non-safety related containment fan coolers used in normal operation are isolated upon a containment isolation signal. The bounding WGOTHIC analysis reported in item a) above has been performed assuming that fan coolers operate after 10 minutes.

c) A NOTRUMP analysis for the DEDVI case using 22 psia containment pressure has been performed, consistent with the WGOTHIC case in Attachment A with NRC assumptions. Table 1 and Figures 1 to 20 provide the results of this NOTRUMP analysis. The AP1000 DCD will be revised to replace the existing DEDVI 25 psia case with the DEDVI 22 psia case shown here. In AP1000 DCD Revision 3 a NOTRUMP case was presented with a containment pressure of 14.7 psia. This analysis is still a valid analysis for the AP1000 and the AP1000 DCD will be revised to re-insert this case.

A WCOBRA/TRAC long term cooling analysis has been performed assuming a containment pressure of 14.7 psia to show that successful long term core cooling is not dependent on elevated containment pressure. The limiting DCD case of a DEDVI break located in the PXS "B" room has been analyzed in the window mode at the most severe time, the time of switchover to containment recirculation. The results of this analysis are given in Figures WCT-1 to WCT-14. Following the problem restart at 6500 seconds, the initial 700 second portion of the WCOBRA/TRAC window establishes the quasi-steady state cooling condition associated with the 14.7 psia containment pressure. Once the 14.7

psia containment boundary pressure quasi-steady state condition is established at 7200 seconds, the window is executed for 2400 seconds to demonstrate adequate ECCS performance. The AP1000 DCD will be revised to add this WCOBRA/TRAC analysis.

Overall Assessment

Westinghouse has performed a considerable effort to evaluate and defend the analysis of minimum possible containment pressure versus time. While containment backpressure provides a positive impact on SBLOCA, ECCS performance analysis performed at a containment pressure of 14.7 psia with NOTRUMP for the short term performance and with WCOBRA/TRAC for the long term cooling show acceptable performance with the containment at atmospheric pressure.

Table-1 DEDVI 22.0 PSI Containment Pressure

Event	Time
Break opens	0.0
Reactor trip signal	13.1
Steam turbine stop valves close	19.1
"S" signal	18.5
Main feed isolation valves begin to close	20.5
Reactor coolant pumps start to coast down	24.5
ADS Stage 1	182.4
ADS Stage 2	252.4
Intact accumulator injection starts	254
ADS Stage 3	372.4
ADS Stage 4	492.4
Intact accumulator empties	600.0
Intact loop IRWST injection starts*	1440
Intact loop core makeup tank empties	2230

Note:
*Continuous injection period

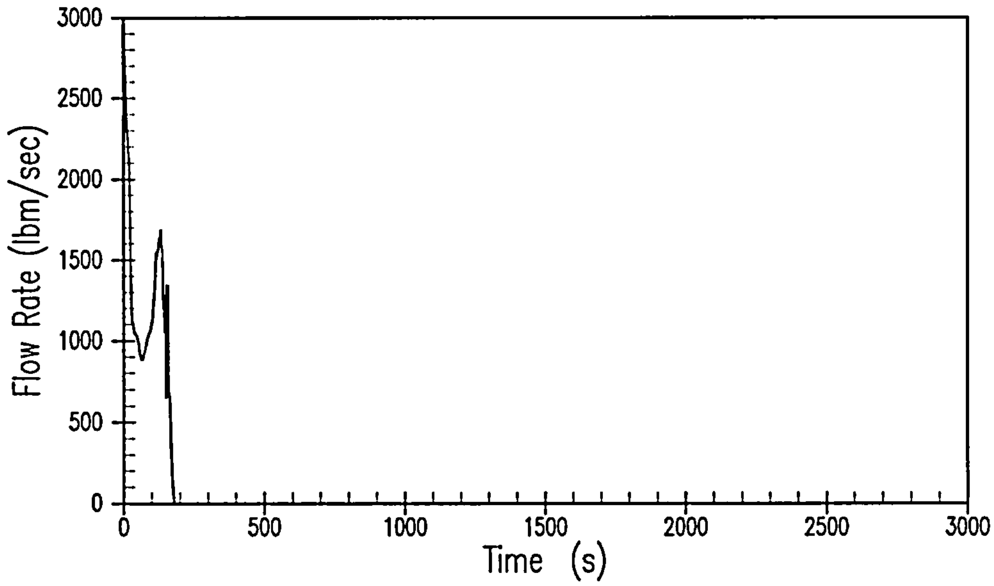


Figure-1 DEDVI 22.0 PSI – Vessel Side Liquid Break Discharge

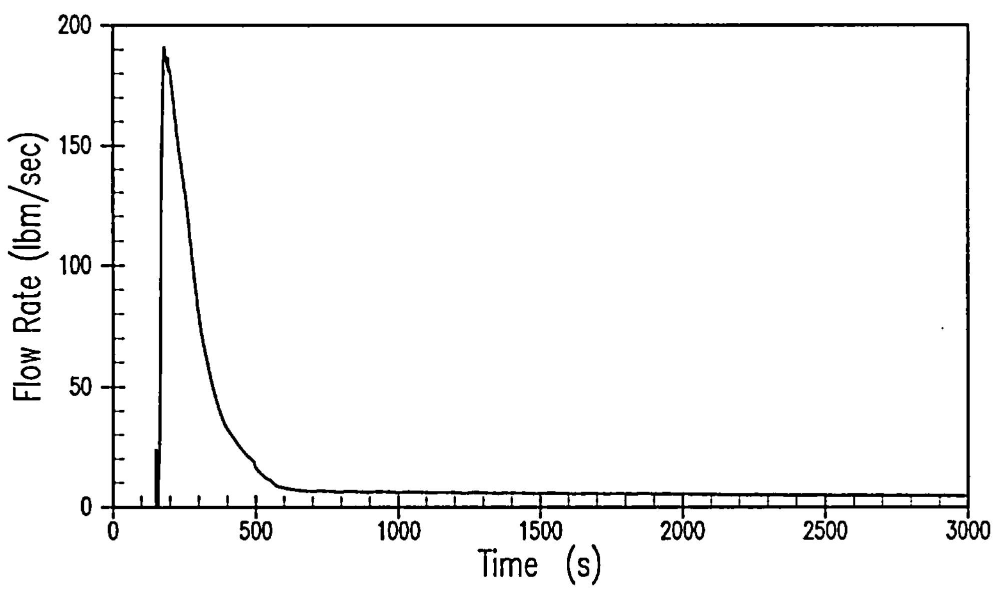


Figure-2 DEDVI 22.0 PSI – Vessel Side Vapor Break Discharge

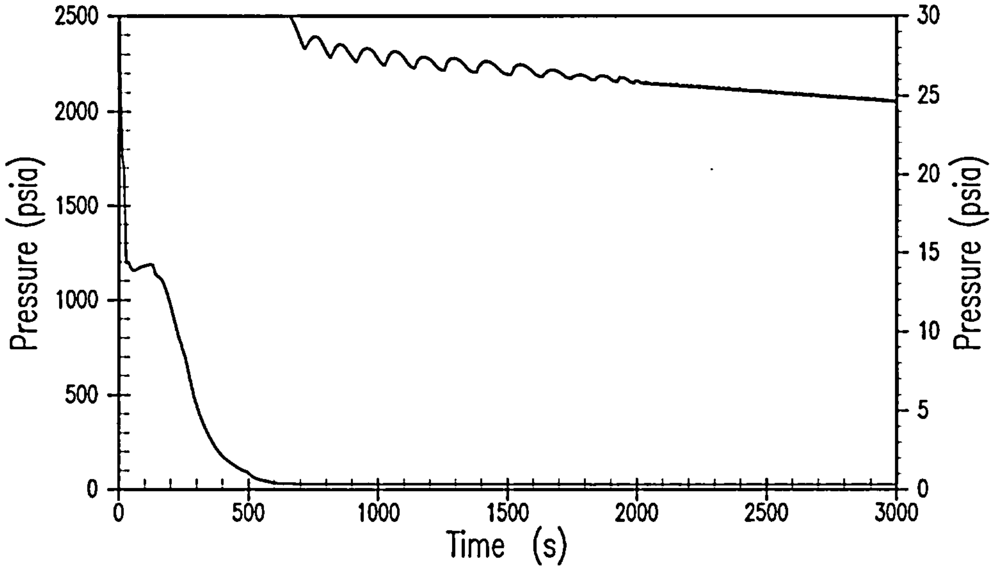


Figure-3 DEDVI 22.0 PSI – RCS Pressure

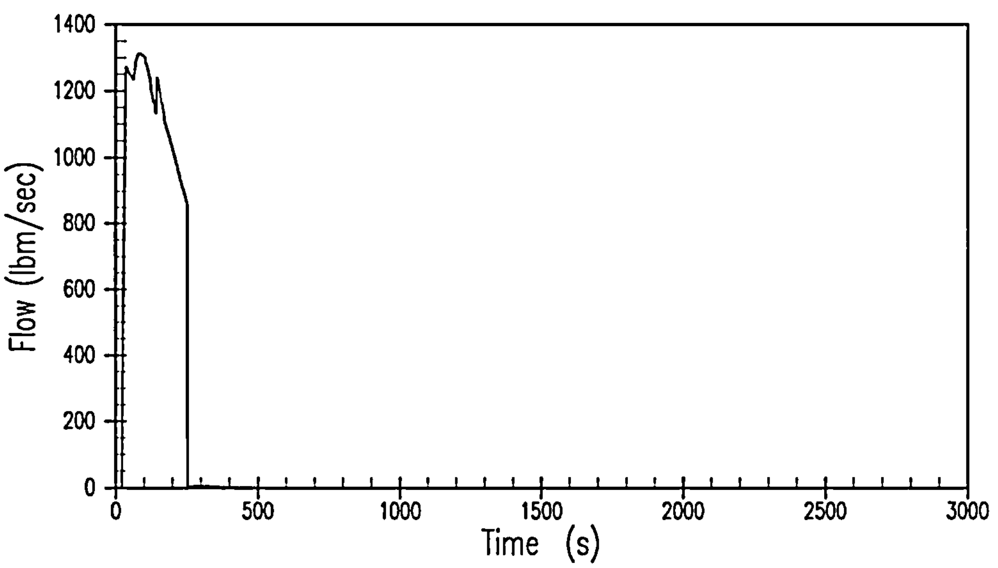


Figure-4 DEDVI 22.0 PSI – Broken CMT Injection Rate

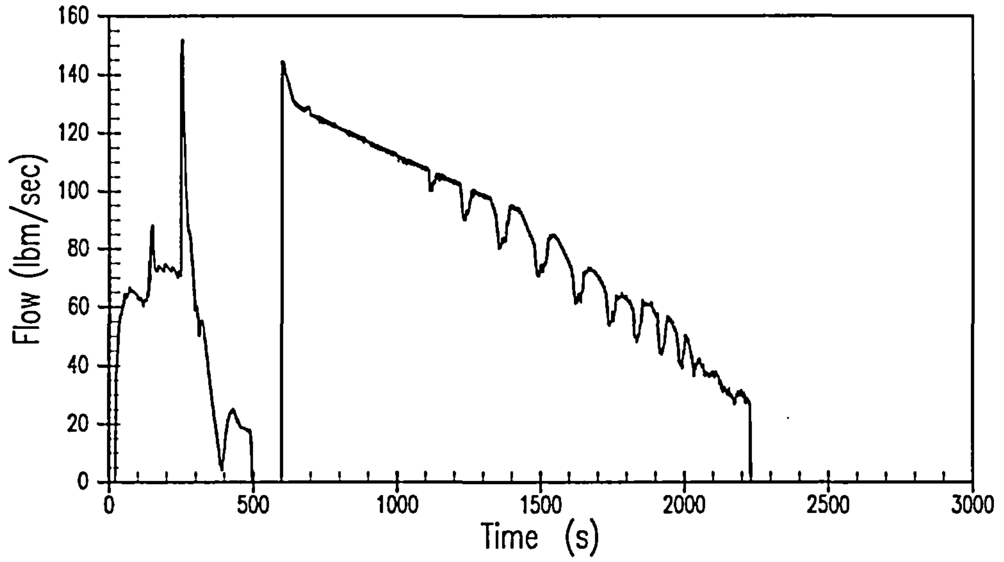


Figure-5 **DEDVI 22.0 PSI – Intact CMT Injection Rate**

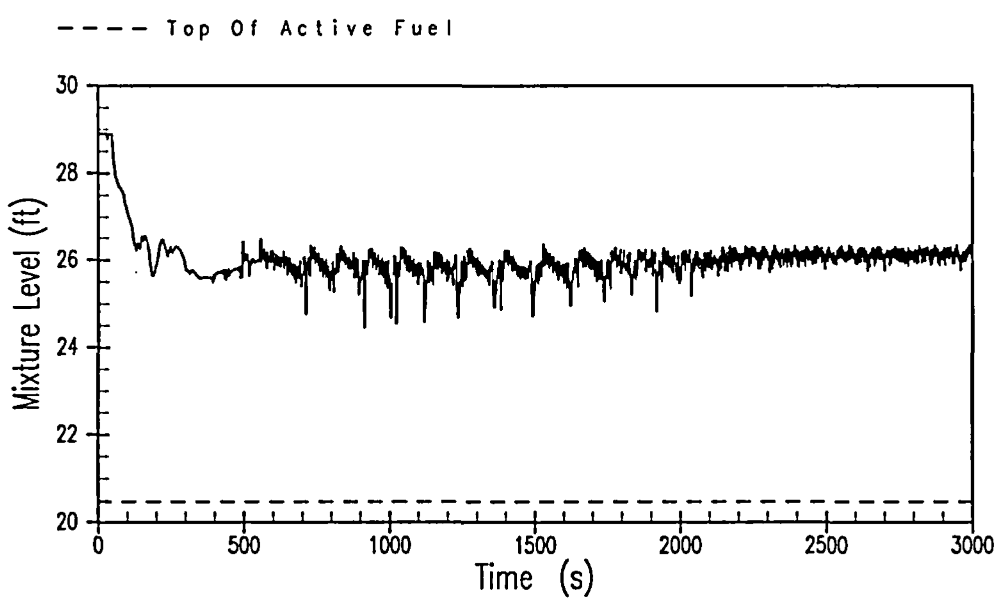


Figure-6 **DEDVI 22.0 PSI – Core/Upper Plenum Mixture Level**

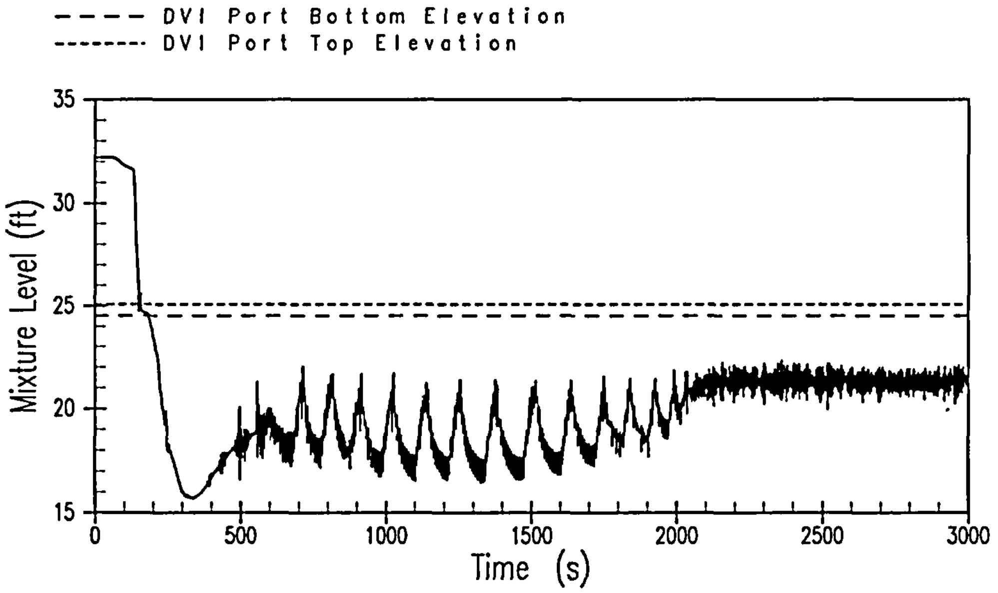


Figure-7 **DEDVI 22.0 PSI – Downcomer Mixture Level**

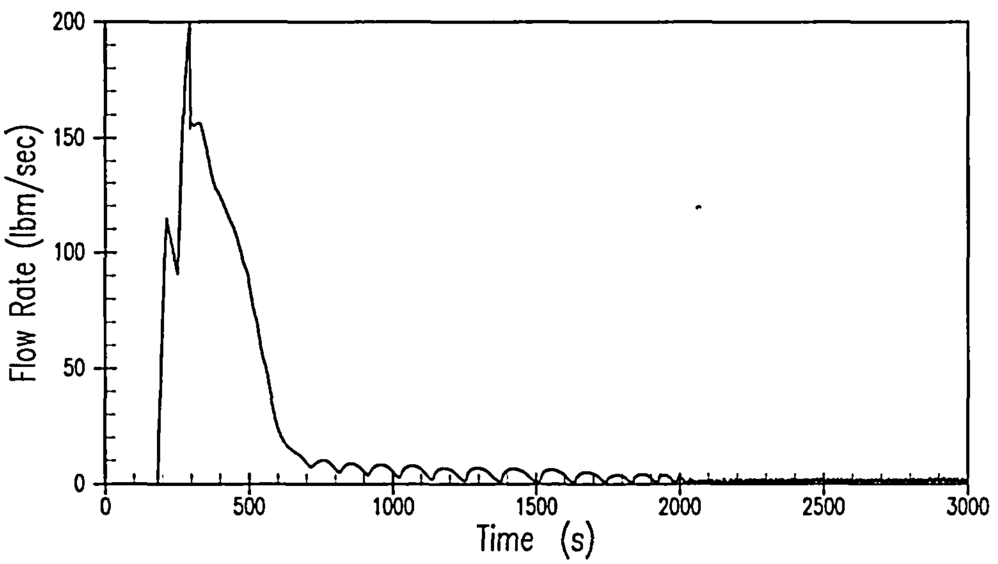


Figure-8 **DEDVI 22.0 PSI – ADS 1-3 Vapor Discharge**

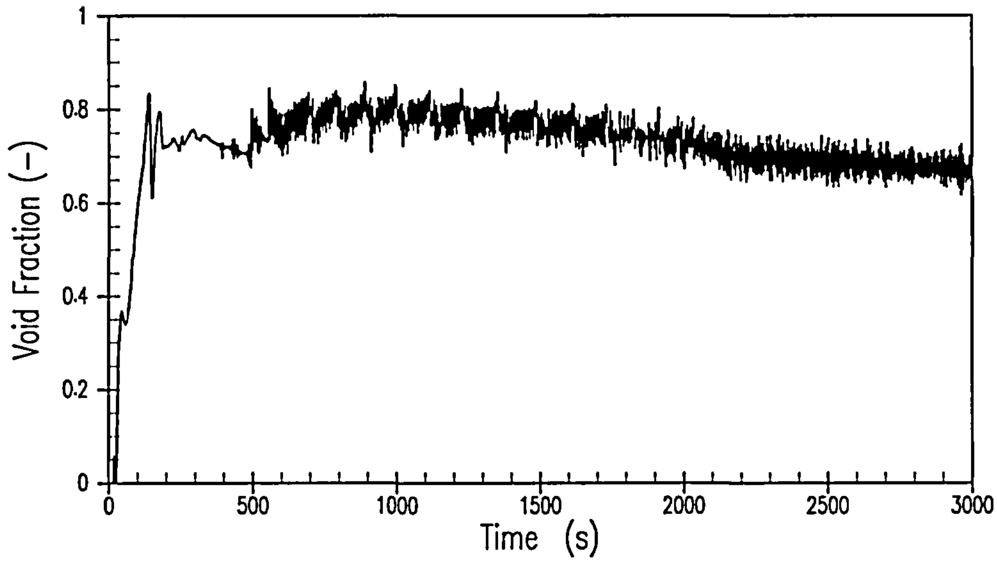


Figure-9 DEDVI 22.0 PSI – Core Exit Void Fraction

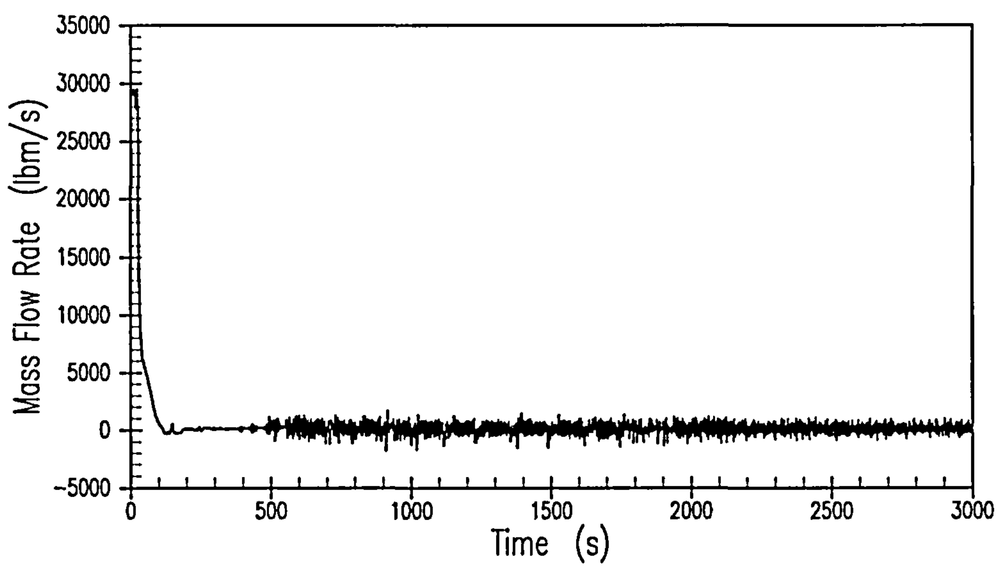


Figure-10 DEDVI 22.0 PSI – Core Exit Liquid Flow Rate

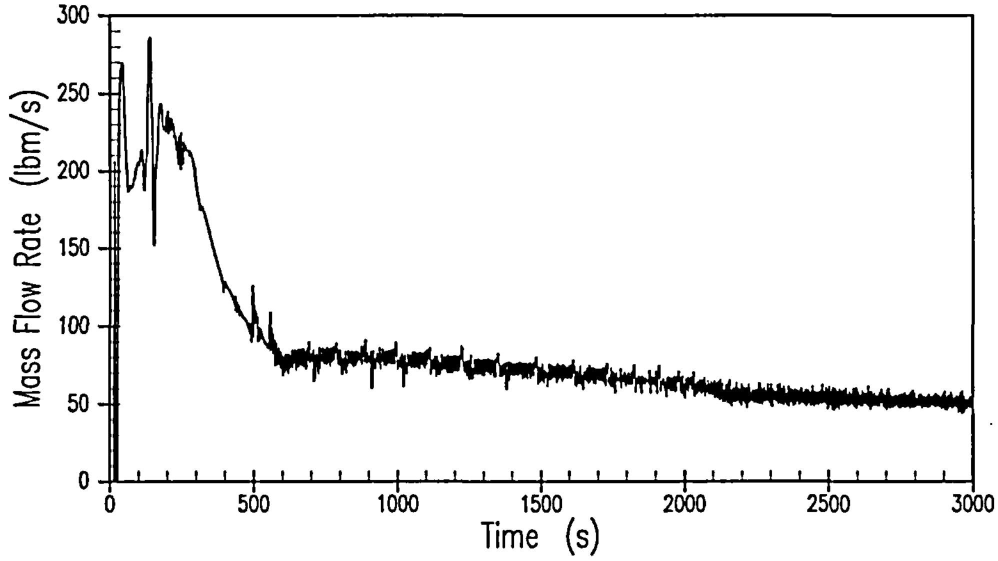


Figure-11 DEDVI 22.0 PSI – Core Exit Vapor Flow Rate

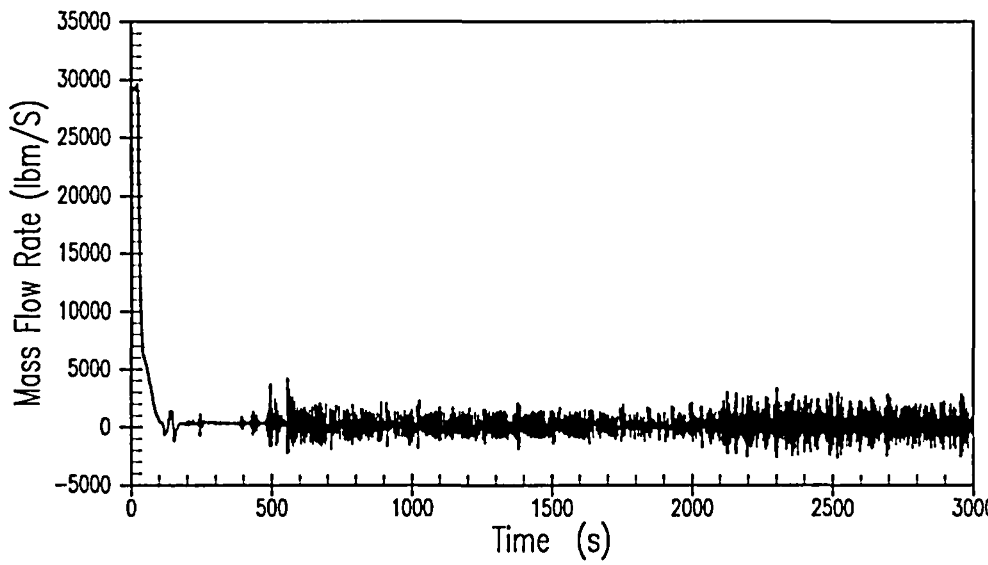


Figure-12 DEDVI 22.0 PSI – Lower Plenum to Core Flow Rate

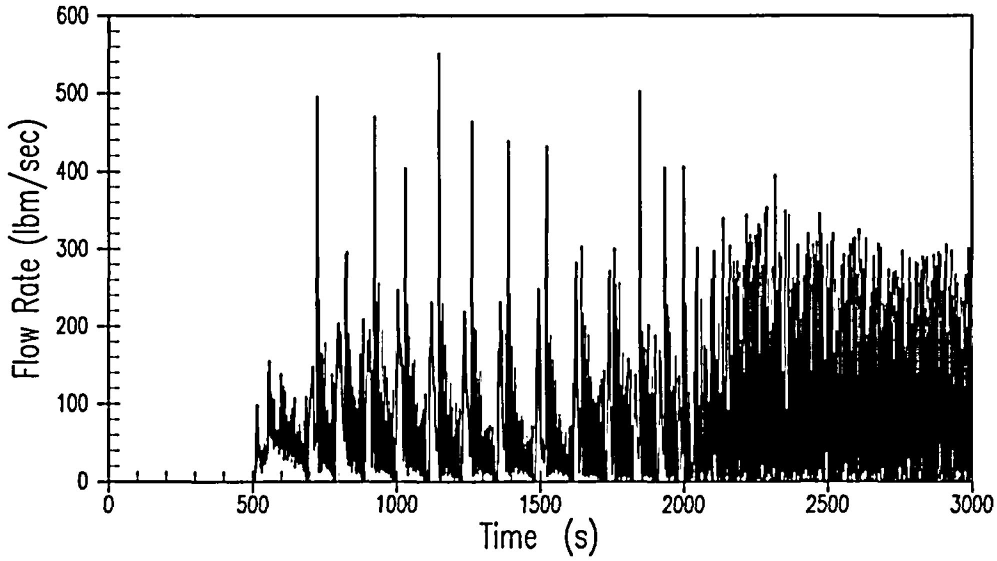


Figure-13 DEDVI 22.0 PSI – ADS-4 Liquid Discharge

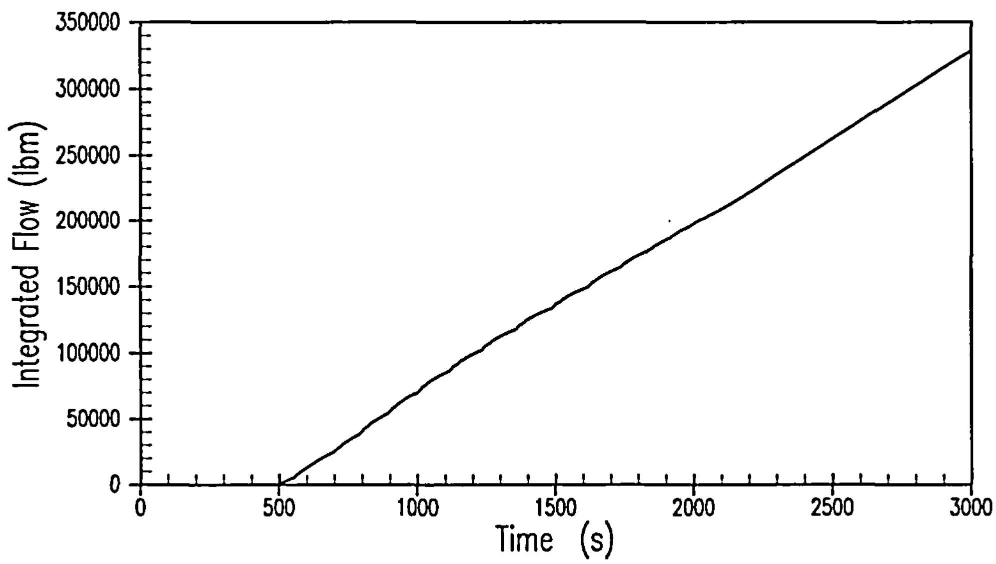


Figure-14 DEDVI 22.0 PSI – ADS-4 Integrated Discharge

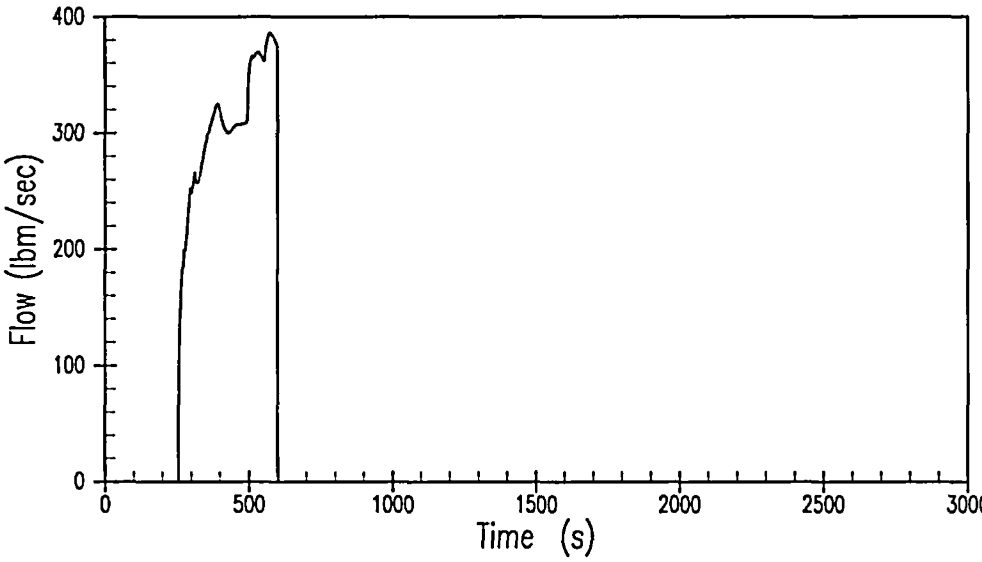


Figure-15 DEDVI 22.0 PSI – Intact Accumulator Flow Rate

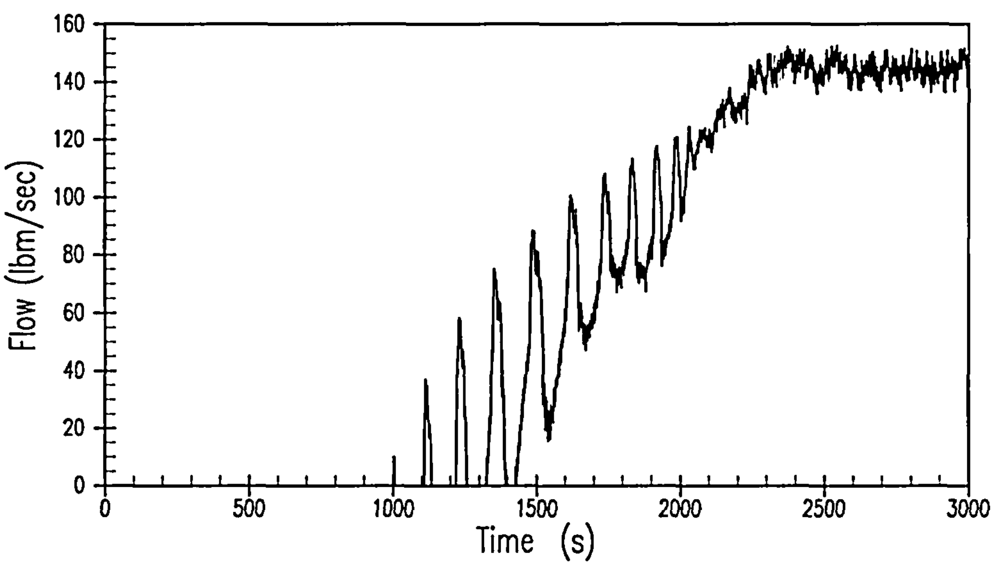


Figure-16 DEDVI 22.0 PSI – Intact IRWST Injection Rate

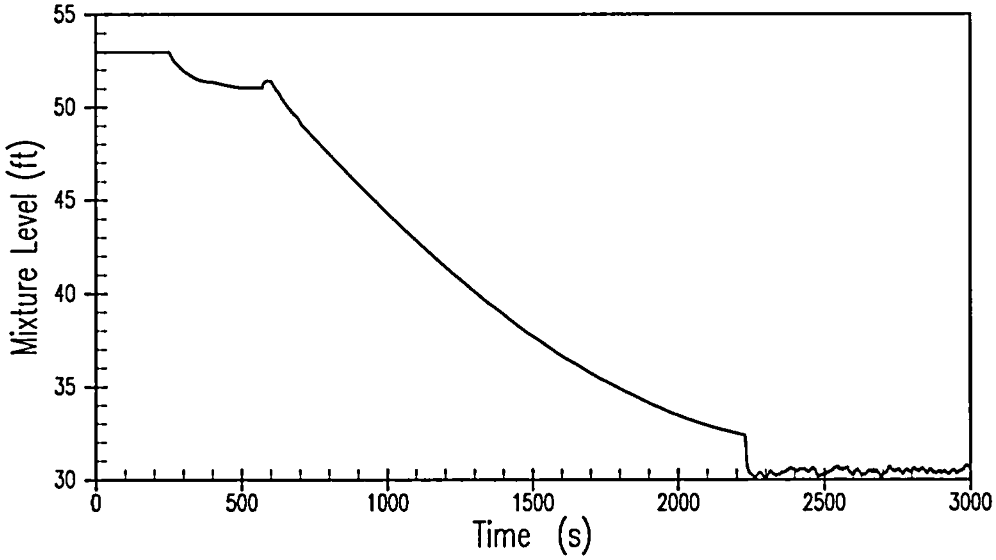


Figure-17 DEDVI 22.0 PSI – Intact CMT Mixture Level

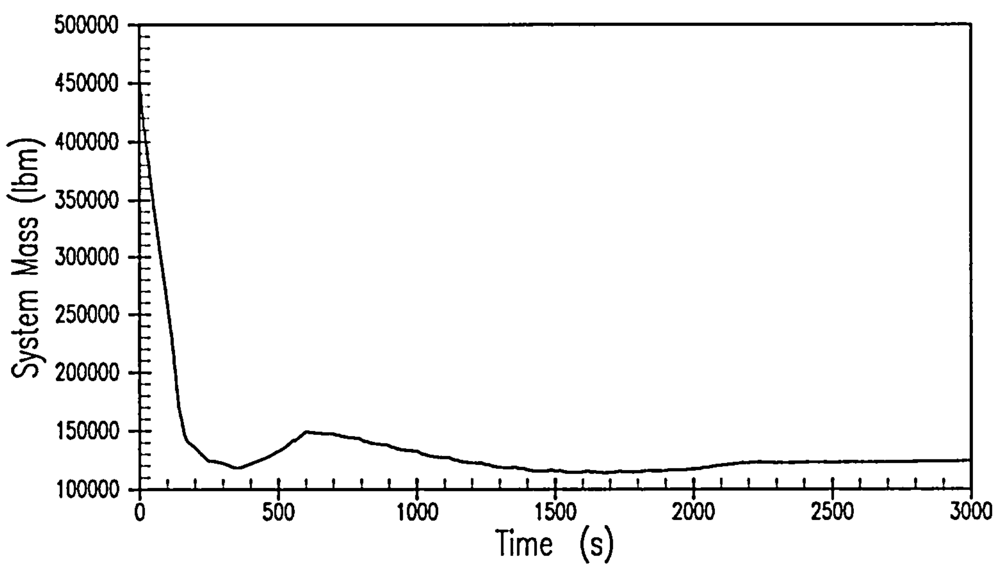


Figure-18 DEDVI 22.0 PSI – RCS System Inventory

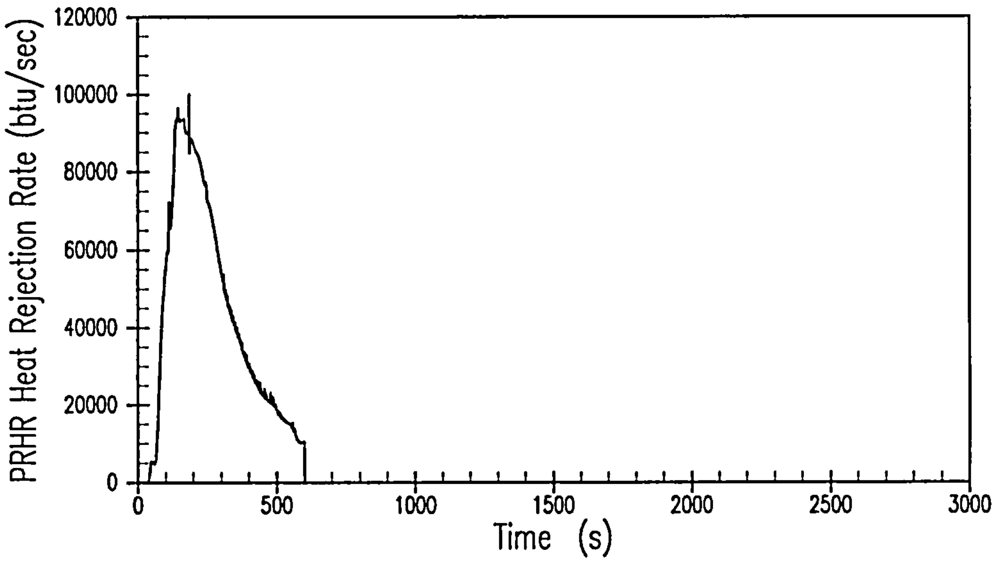


Figure-19 DEDVI 22.0 PSI – PRHR Heat Removal Rate

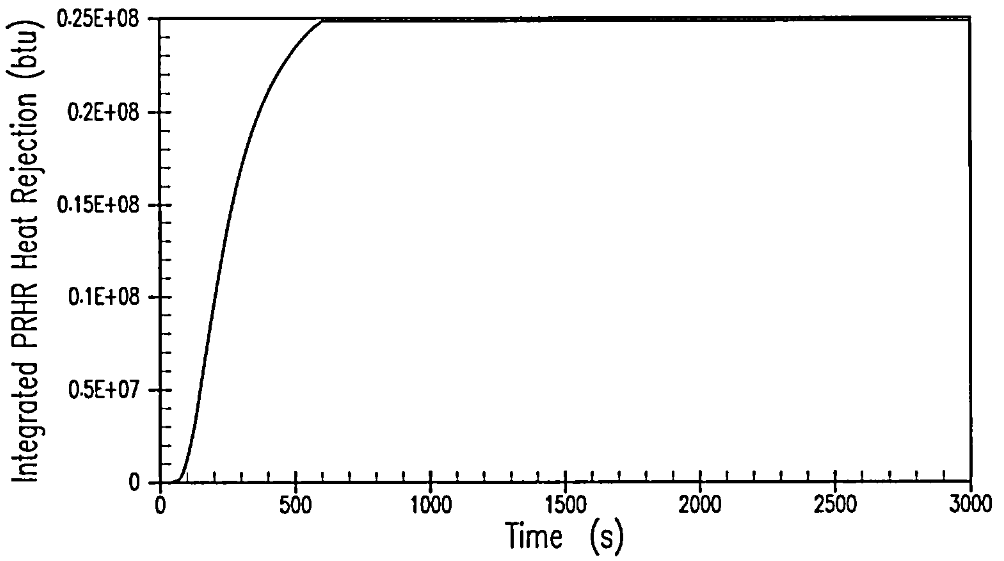
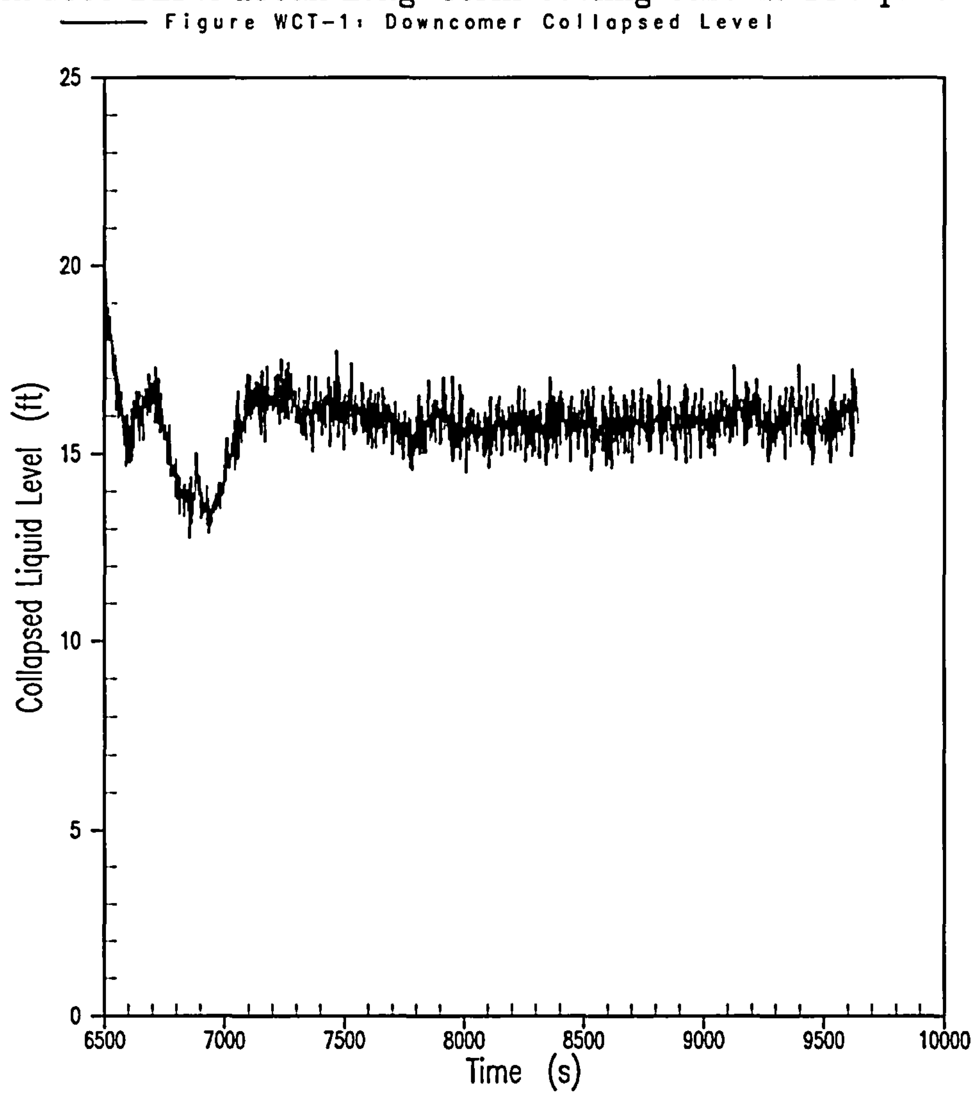
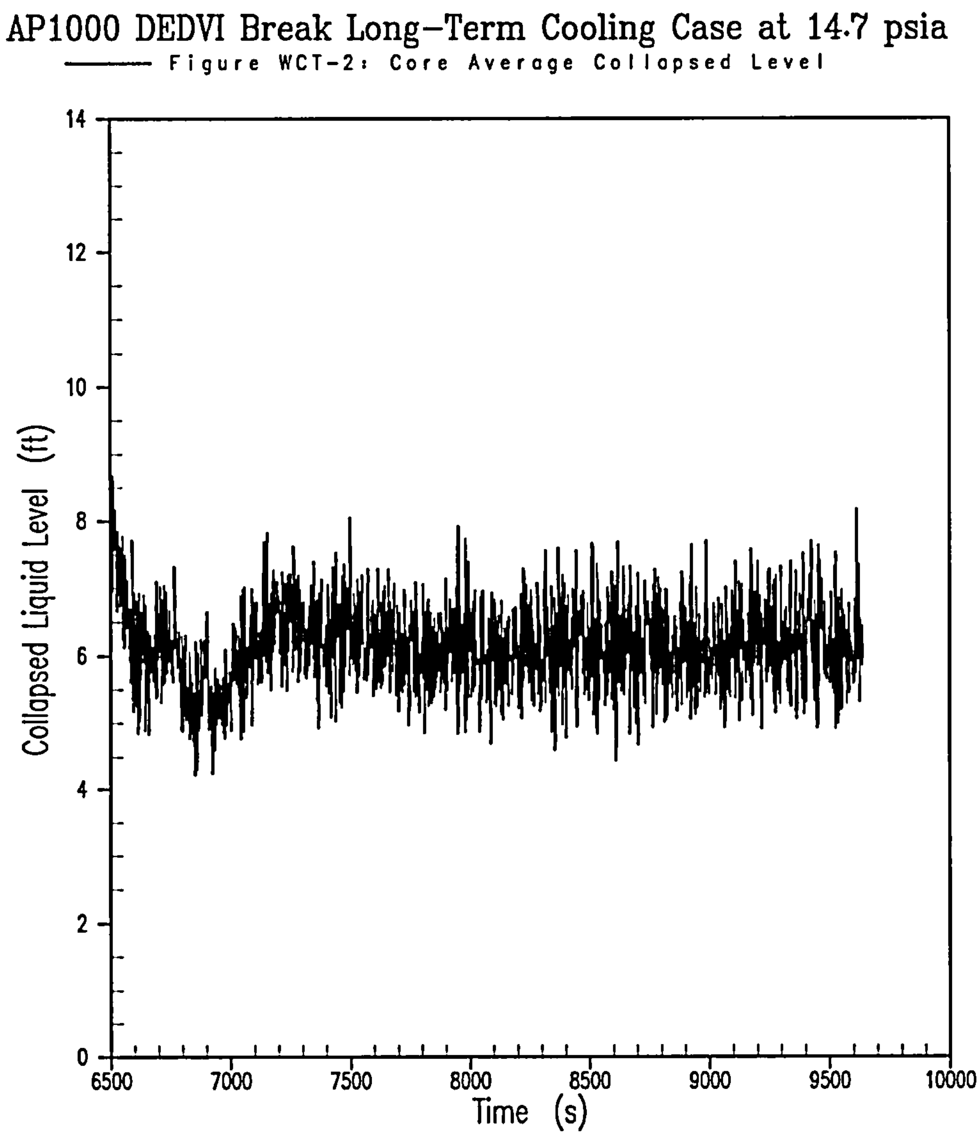


Figure-20 DEDVI 22.0 PSI – Integrated PRHR Heat Removal

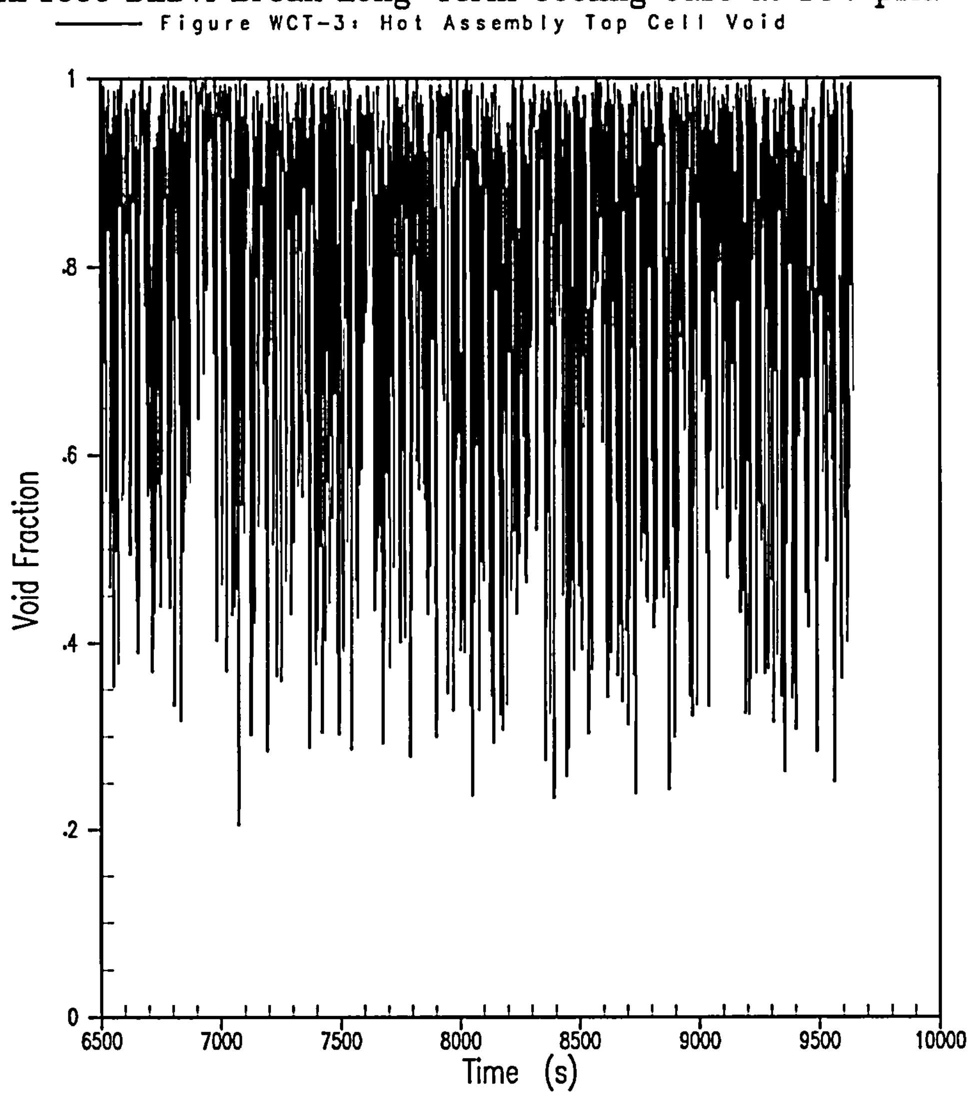
AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia

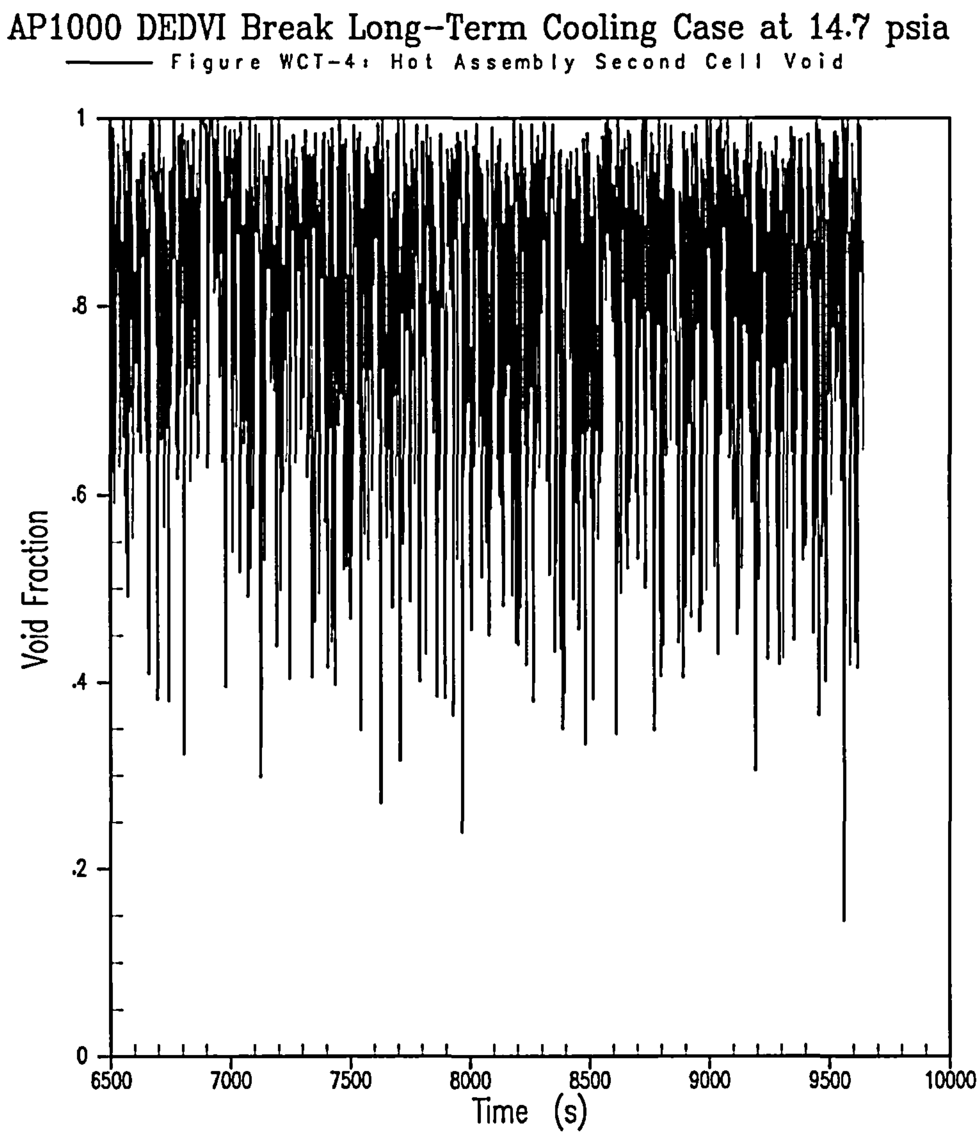


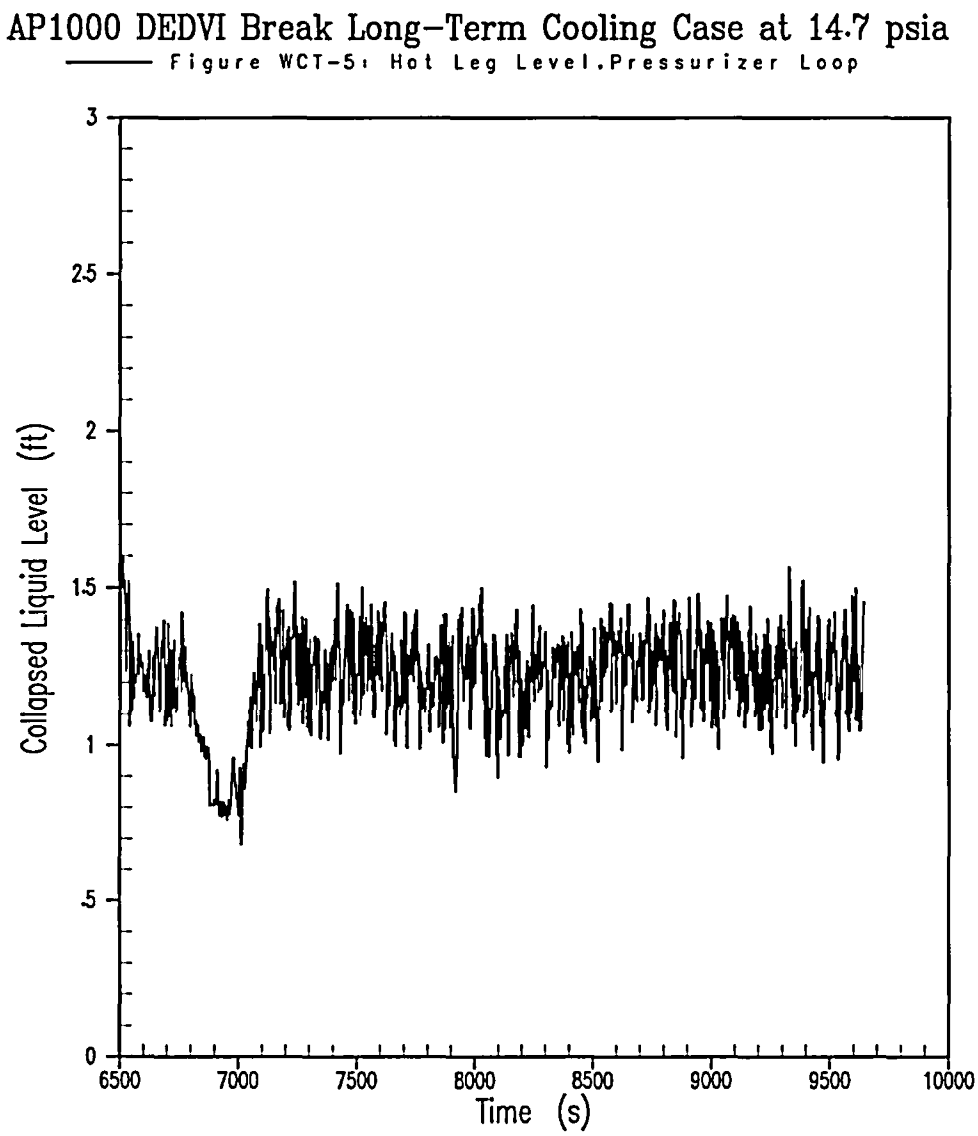


AP1000 DEDVI Break Long–Term Cooling Case at 14.7 psia
Figure WCT-2: Core Average Collapsed Level
14
12
10
8
6
4
2
0
Collapsed Liquid Level (ft)
6500
7000
7500
8000
8500
9000
9500
10000
Time (s)

AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia

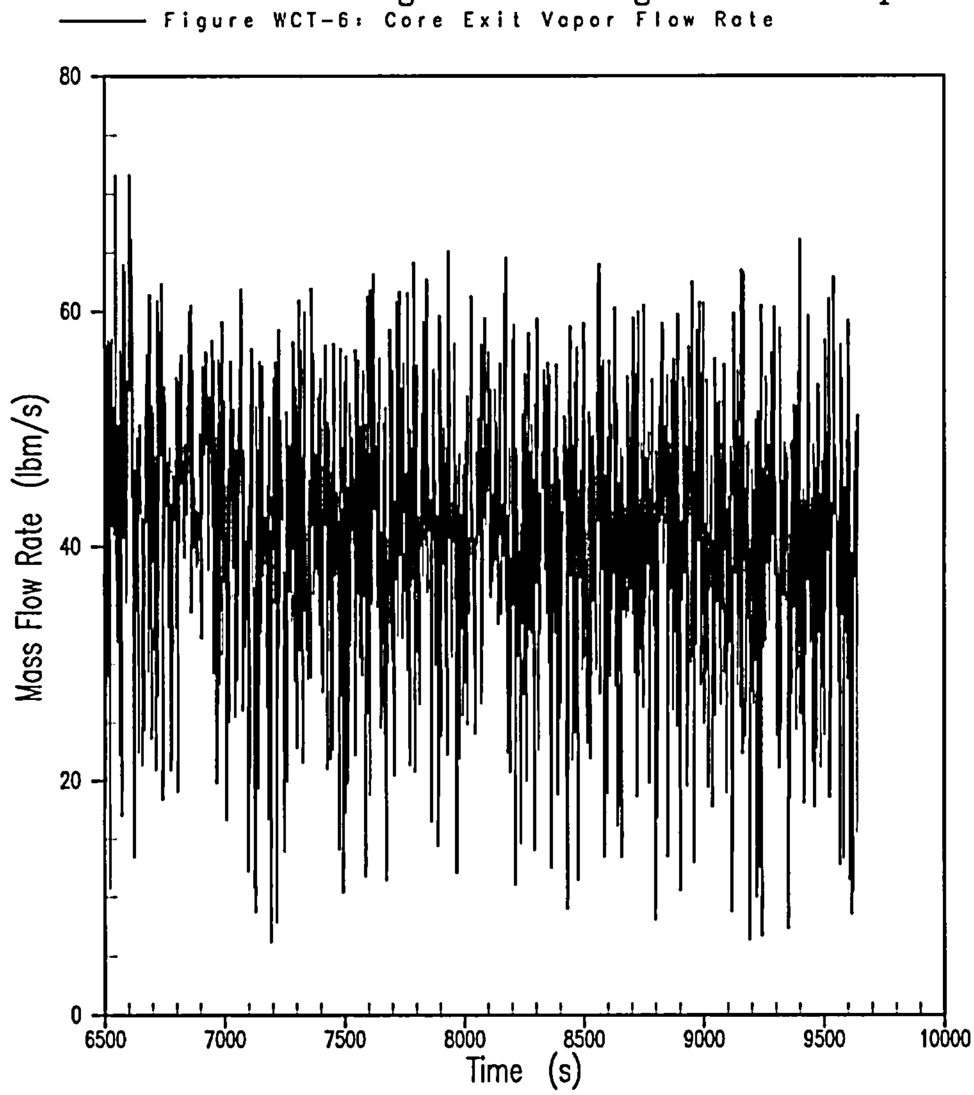


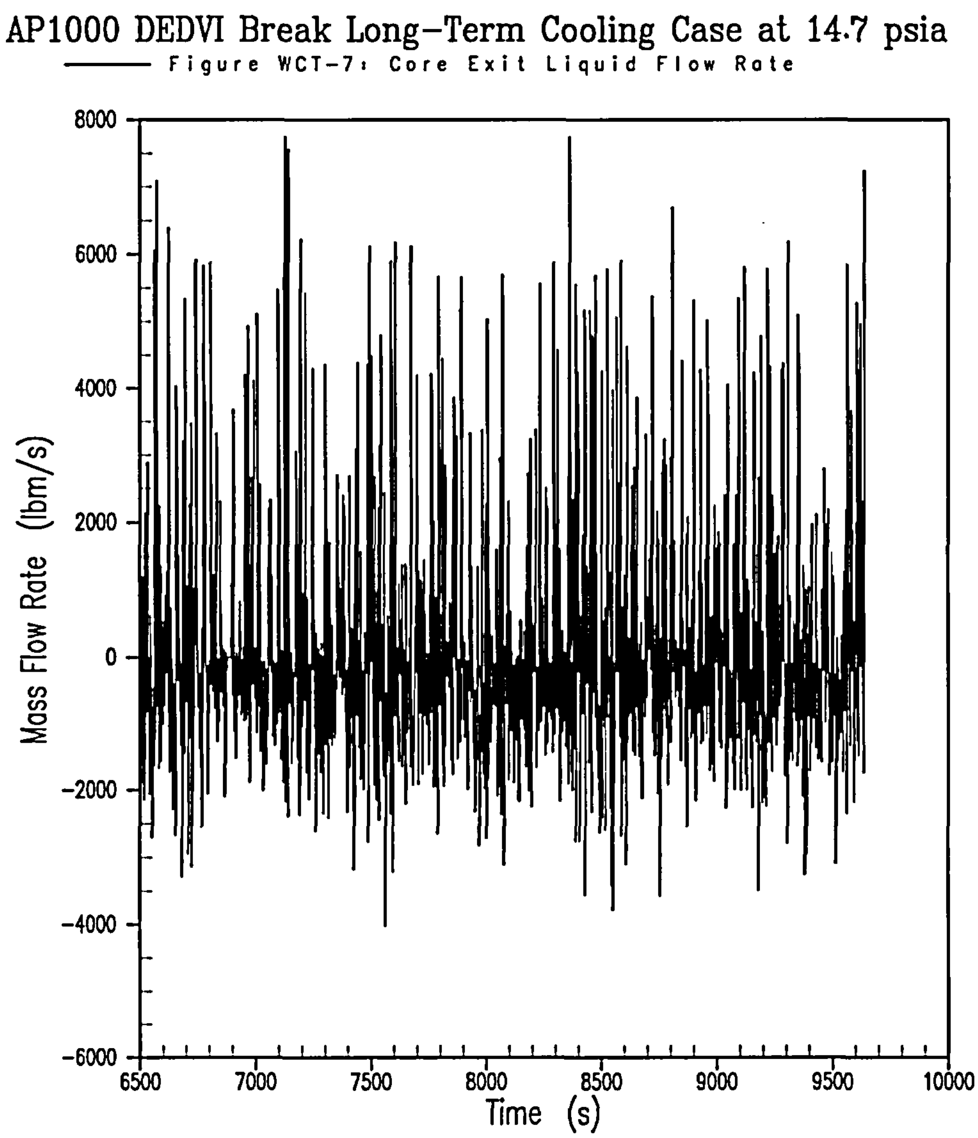

AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia
Figure WCT-4: Hot Assembly Second Cell Void
Void Fraction
1
.8
.6
.4
.2
0
6500
7000
7500
8000
8500
9000
9500
10000
Time (s)



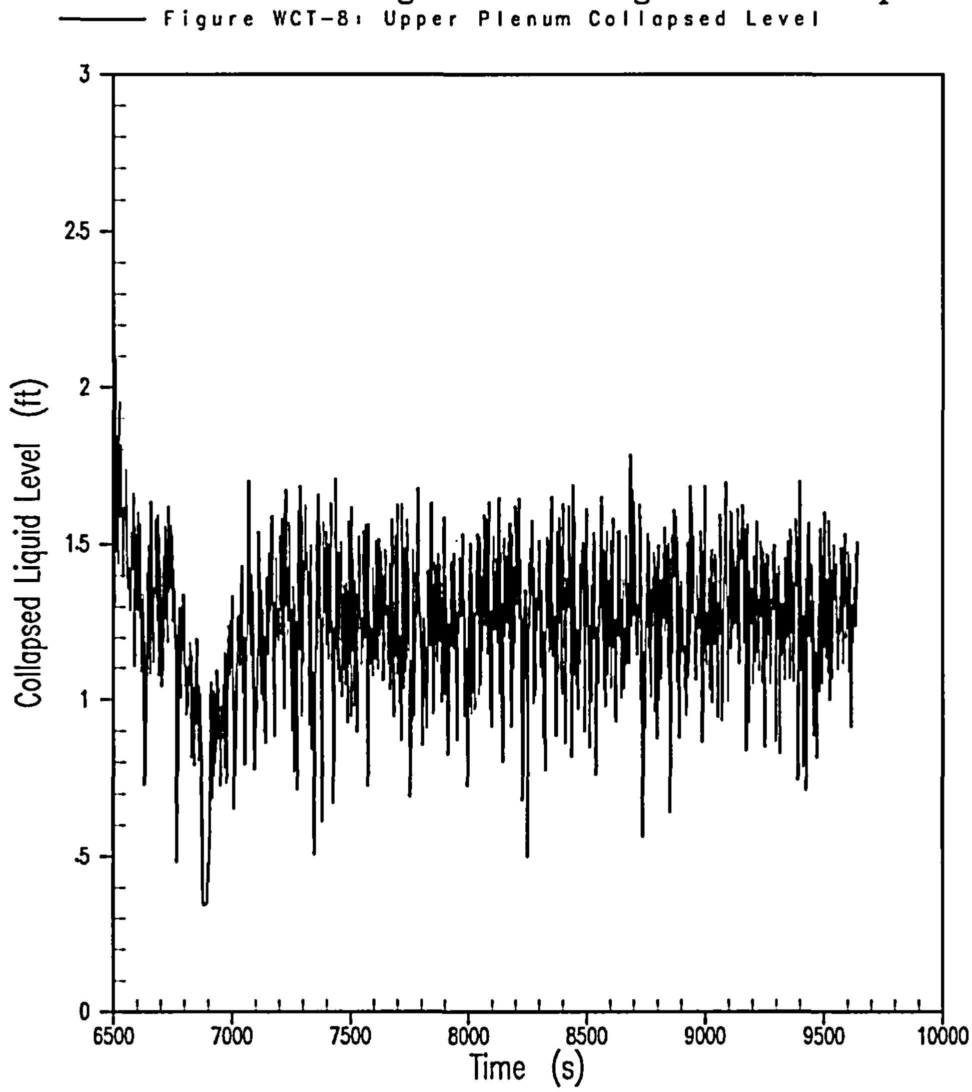
AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia
Figure WCT-5: Hot Leg Level.Pressurizer Loop
Collapsed Liquid Level (ft)
3
2.5
2
1.5
1
.5
0
6500
7000
7500
8000
8500
9000
9500
10000
Time (s)

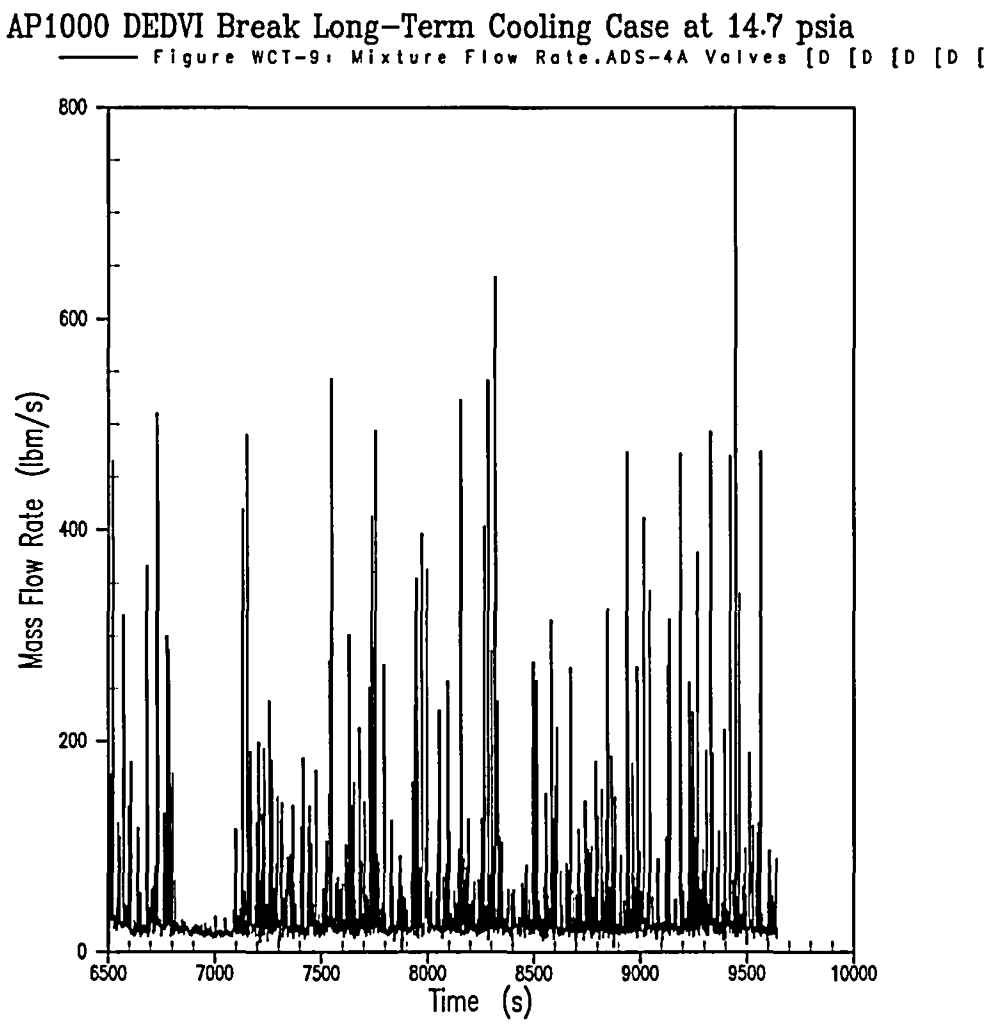
AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia




AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia
Figure WCT-7: Core Exit Liquid Flow Rate
8000
6000
4000
2000
0
-2000
-4000
-6000
Mass Flow Rate (lbm/s)
6500
7000
7500
8000
8500
9000
9500
10000
Time (s)

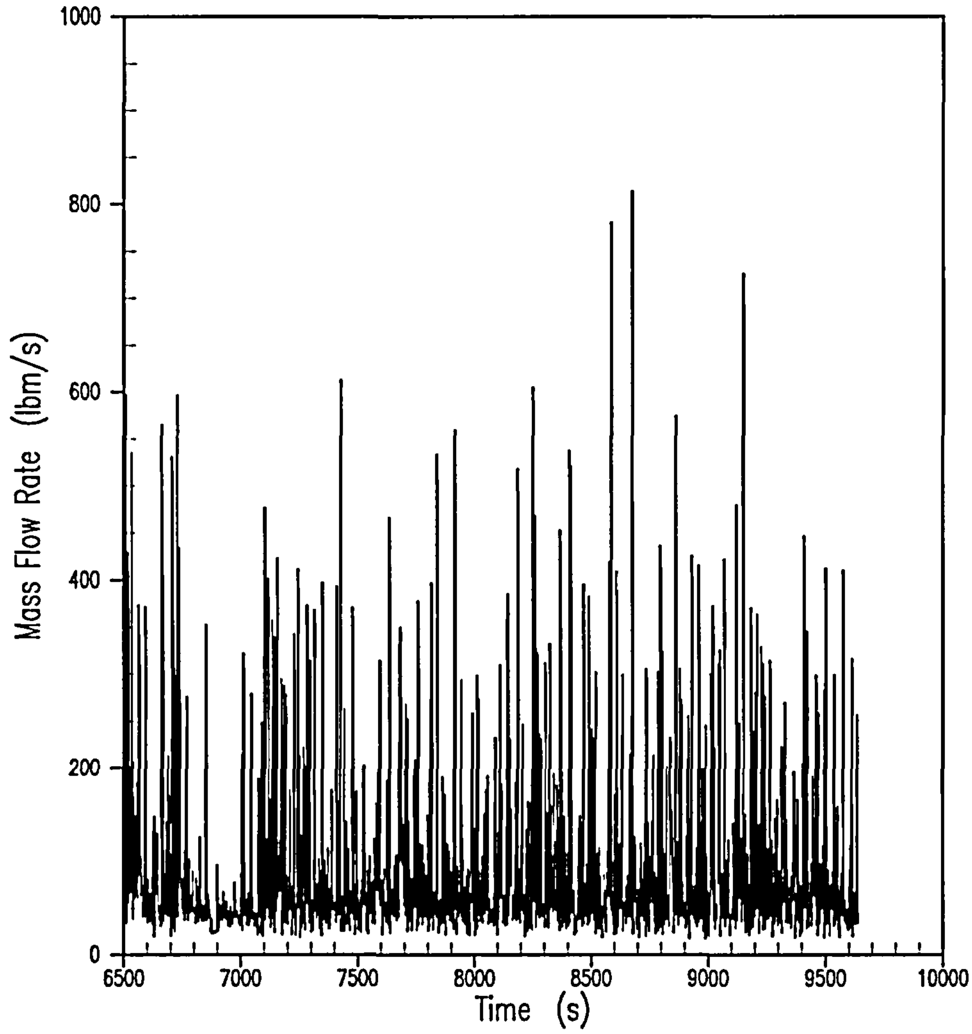
AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia




AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia
Figure WCT-9: Mixture Flow Rate.ADS-4A Valves [D [D [D [D [
Mass Flow Rate (lbm/s)
800
600
400
200
0
6500
7000
7500
8000
8500
9000
9500
10000
Time (s)

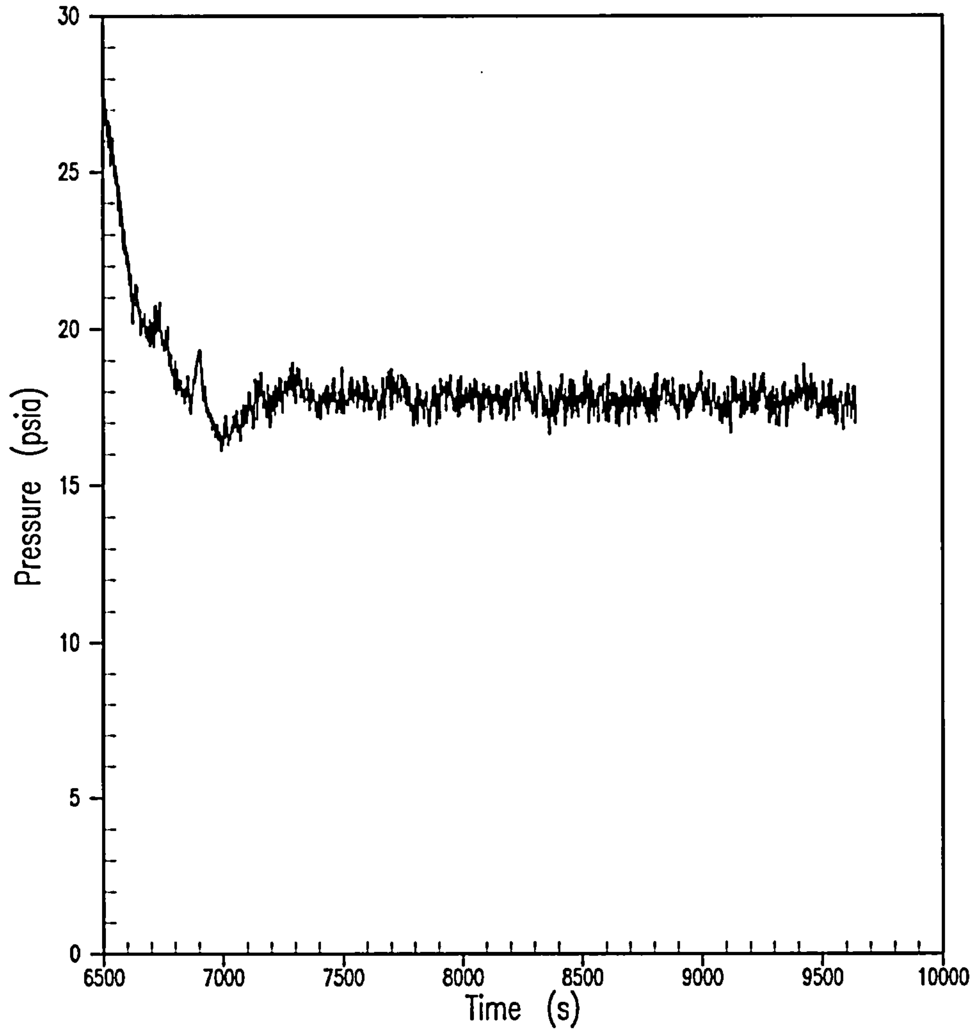
AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia

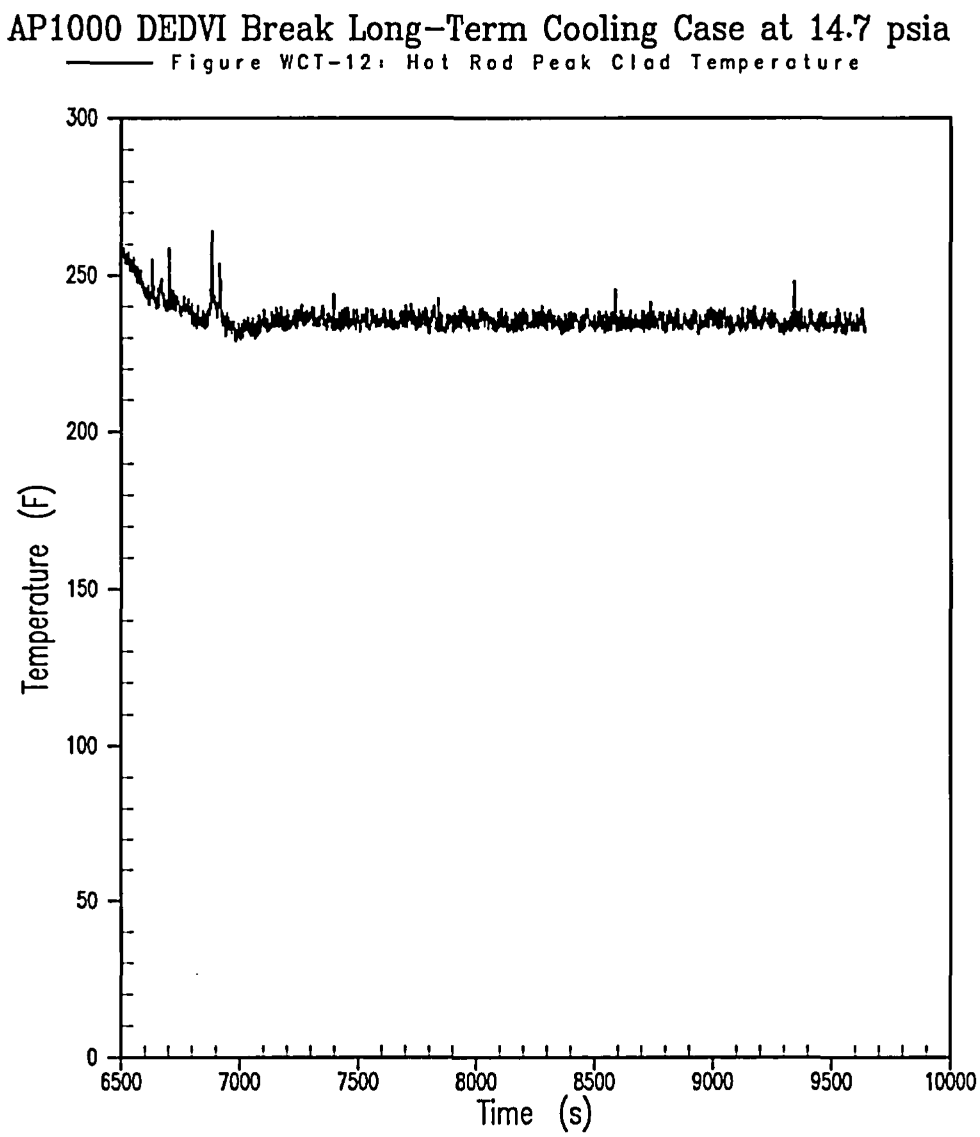
Figure WCT-10: Mixture Flow Rate.ADS-4B Valves



AP1000 DEDVI Break Long–Term Cooling Case at 14.7 psia

——— Figure WCT-11: Upper Plenum Pressure

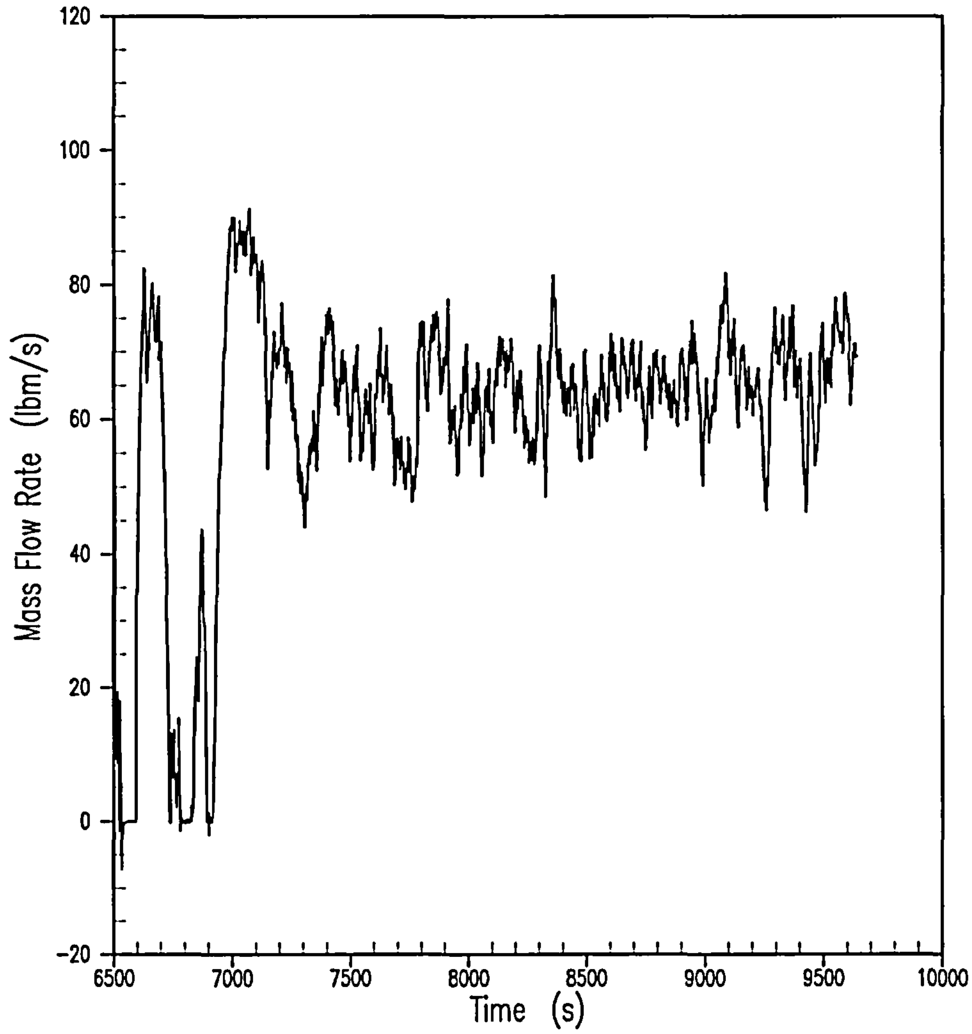


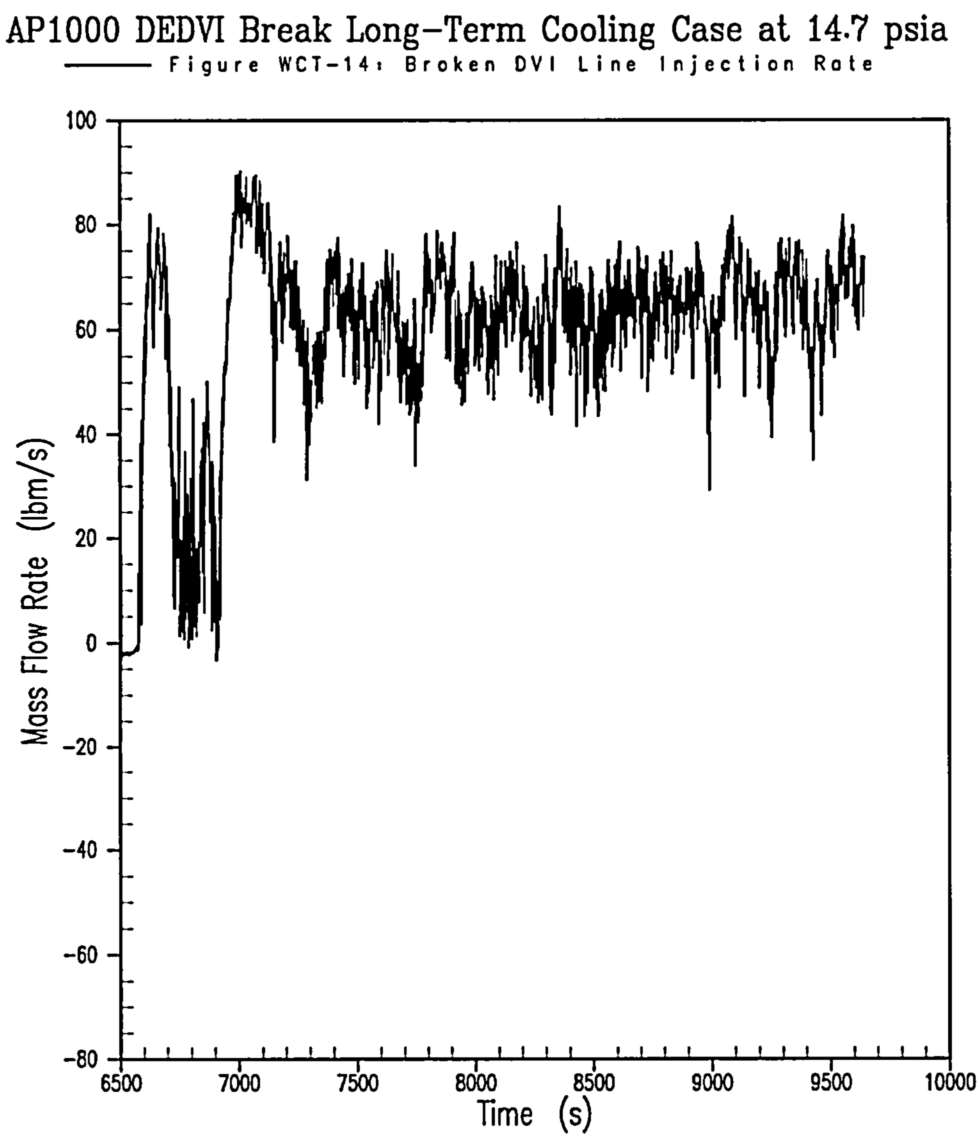


AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia
Figure WCT-12: Hot Rod Peak Clad Temperature
Temperature (F)
300
250
200
150
100
50
0
6500
7000
7500
8000
8500
9000
9500
10000
Time (s)

AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia

Figure WCT-13: Intact DVI Line Injection Rate





AP1000 DEDVI Break Long-Term Cooling Case at 14.7 psia
Figure WCT-14: Broken DVI Line Injection Rate
Mass Flow Rate (lbm/s)
Time (s)
100
80
60
40
20
0
-20
-40
-60
-80
6500
7000
7500
8000
8500
9000
9500
10000

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 15.2.7-1 Item 12 Response Revision 1

Original RAI Number(s): None

Summary of Issue:

Please provide the following information regarding the boron concentration analysis given in DCP/NRC1612 entitled "Transmittal of Westinghouse Response to Boron Precipitation during LTC Phase" dated August 15, 2003:

A. ADS-4 quality and liquid mass flow rate versus time used to generate Fig. 5.
B. RCS injection mass flow rate versus time for the analysis in Fig. 5 (CMTs, accumulators, IRWST, and Sump).
C. Core inlet mass flow rate and fluid temperature versus time for Fig. 5
D. A plot of the core exit steaming mass flow rate versus time from Fig. 5.
E. Hot leg void fraction versus time
F. Core exit void fraction versus time

Westinghouse Response (Revision 1):

Attachment A provides Westinghouse response to an item identified in the 12/17/03 Open Item status meeting.

Westinghouse Response (original response):

As discussed in our 8/15/03 response, the T/H analysis that supports the results shown in Fig 5 are taken from two sources. The detailed WCOBRA-TRAC LTC analysis provided in the DCD (subsection 15.6.5.4C) is used for the first 3.5 hours and a simplified hand calculation model is used after that time. Note that the WCOBRA-TRAC analysis covers the limiting time for peak core boron concentration.

The detailed WCOBRA-TRAC LTC analysis results, using the large-LOCA like core noding, are shown in the DCD revision 7 (subsection 15.6.5.4C). This analysis is carried out for 12,500 sec (~3.5 hr), until quasi-steady-state recirculation conditions are established. These results show that the average ADS 4 vent quality increases from below 40% at the start of IRWST injection to below 50% at the end of IRWST injection. The ADS 4 vent quality then trends downward, reaching 43% at 12,500 sec. Attached are plots of the information requested. Note that these plots use smoothing techniques so that it is easy to read the average values. The times start at the beginning of the WCOBRA-TRAC analysis phase which is 2500 sec after the accident; a time of 10,000 sec on one of these plots is 12,500 sec transient time.

The simplified analysis used in the longer-term assumes steady-state conditions, such that the RCS injection, core inlet / outlet and ADS 4 vent flows are all equal. The core inlet temperature vs time is shown on Fig 5 of our 8/15/03 response. The simplified model is applied at fixed times such that the results at any time are independent of the calculations at other times. For the purposes of determining the plant performance after the end of the detailed WCOBRA-TRAC analysis (at 3.5 hr) only a few times are required to follow the slowly changing conditions. The plant conditions change slowly during recirc because the injection supply is essentially constant and the decay heat is slowly decreasing. Note that in the core boron concentration calculations, some margin has been added to the ADS 4 vent qualities; for example at 6 hour the simplified model calculates a ADS 4 vent quality of 34.5%, however 40% is used in the core boron calculations.

The following table provides results from the simplified model:

Times (hr)	ADS 4 Quality (%)	Inject (lb/sec)	ADS 4 Liq (lb/sec)	Core Exit Steam (lb/sec)	Core Exit Void	HL Void
3.0	38.0%	84.9	52.6	31.2	0.997	0.57
6.0	34.5%	89.0	58.3	29.6	0.997	0.56
12.0	26.4%	93.4	68.7	23.5	0.995	0.45

The following table shows the ASD 4 vent quality used in calculating the core boron concentration (listed under "Max") as compared with the results from WCOBRA-TRAC and the simplified model.

Time		ADS 4 Vent Quality		
(hr)	(day)	Max	WCT	Simplified
0	-	60.0%	41%	-
3	-	60.0%	46%	38.0%
12	0.5	29.0%	-	26.4%
24	1.0	21.9%	-	21.9%
-	3.0	14.7%	-	14.7%
-	7.0	10.7%	-	10.7%
-	14.0	7.7%	7.5%	7.7%
-	30.0	5.0%	-	5.0%

Design Control Document (DCD) Revision:

None

PRA Revision:

None

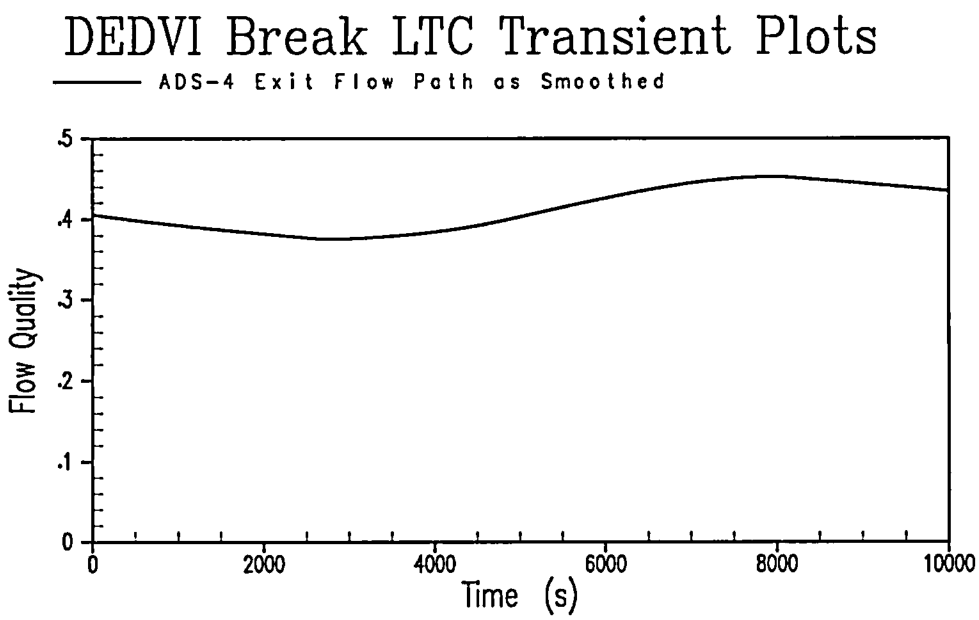


Figure 15.2.7-1-1 ADS 4 Flow Quality

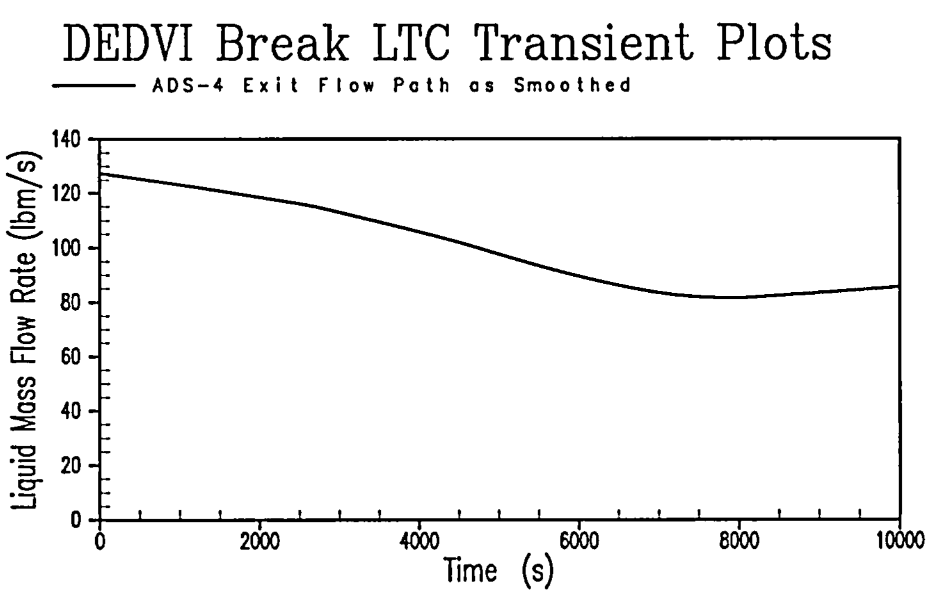


Figure 15.2.7-1-2 ADS-4 Liquid Mass Flow Rate

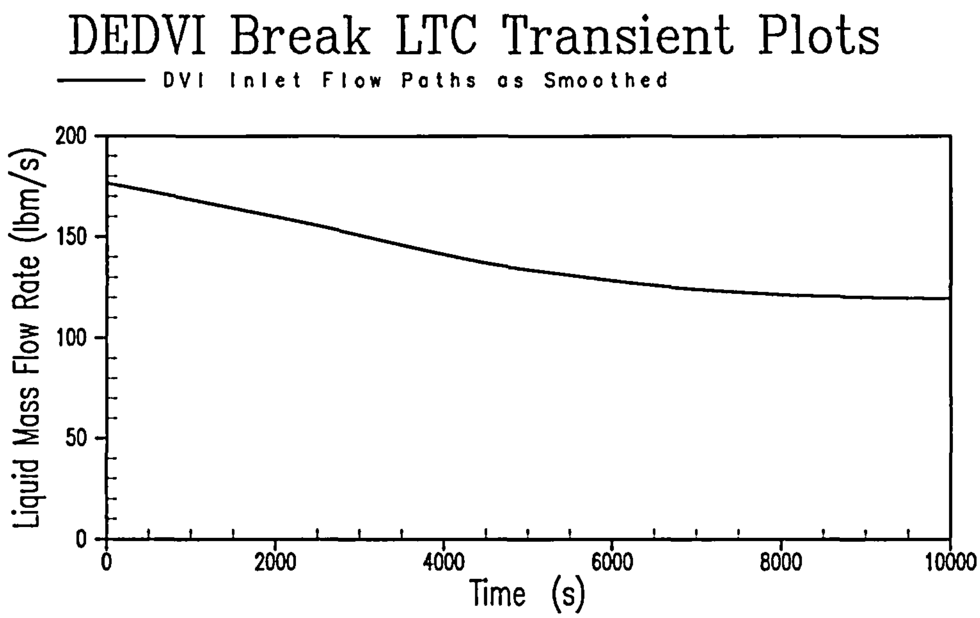


Figure 15.2.7-1-3 DVI Injection Flow

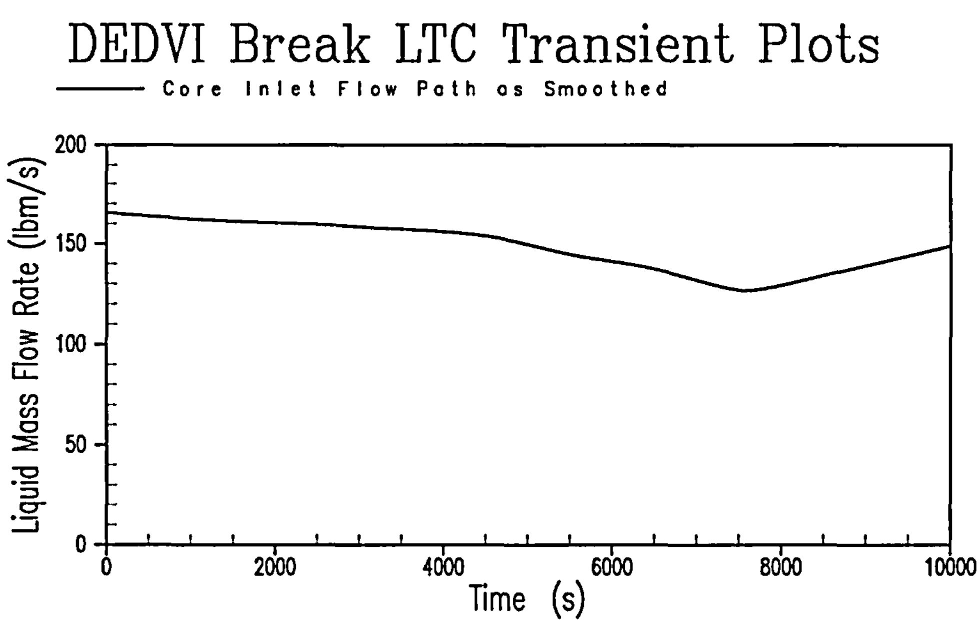


Figure 15.2.7-1-4 Core Inlet Flow

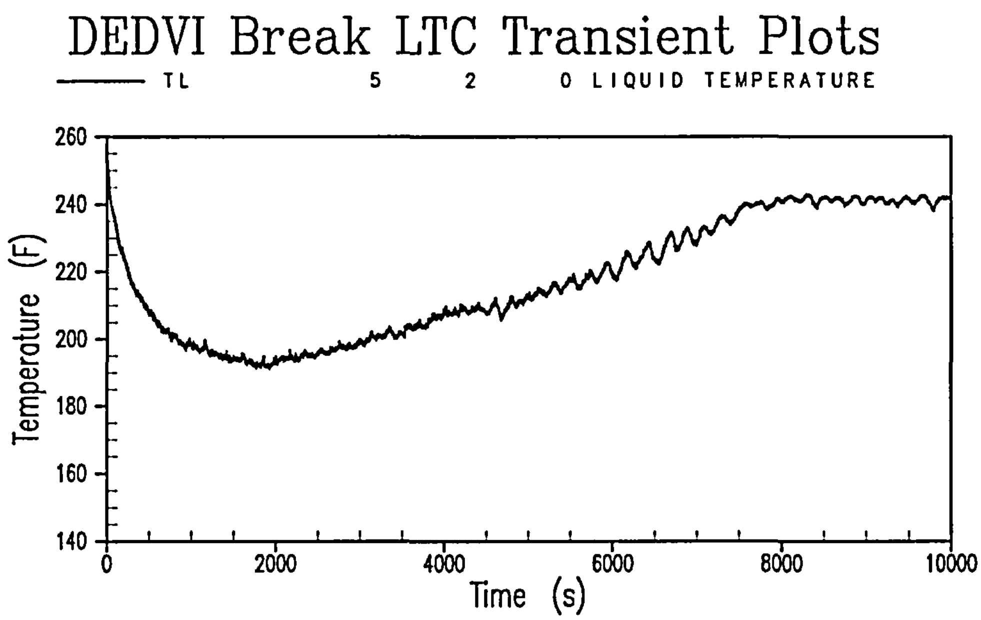


Figure 15.2.7-1-5 Core Inlet Temperature

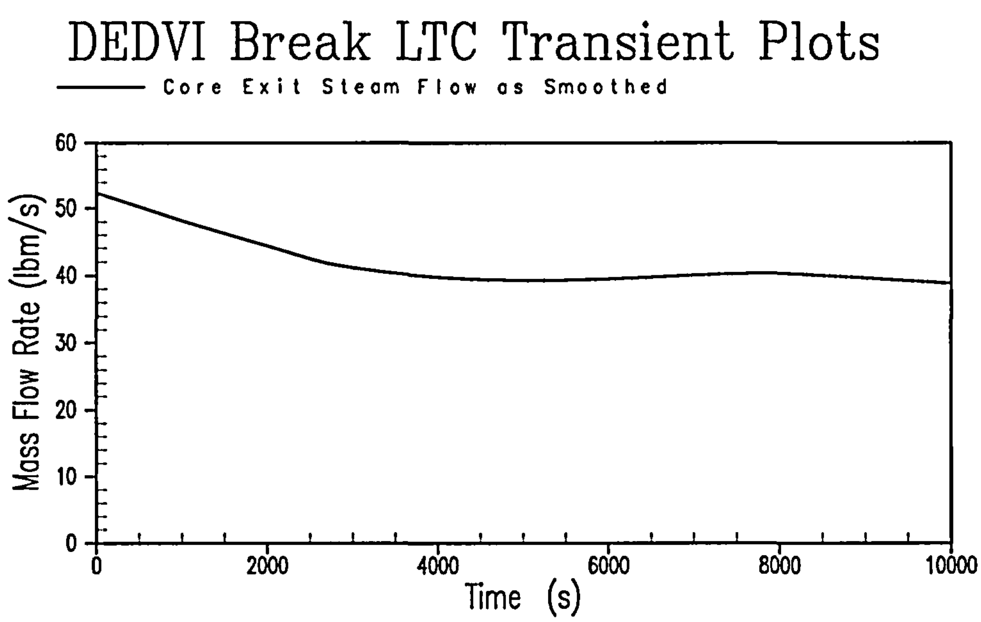


Figure 15.2.7-6 Core Exit Steam Flow

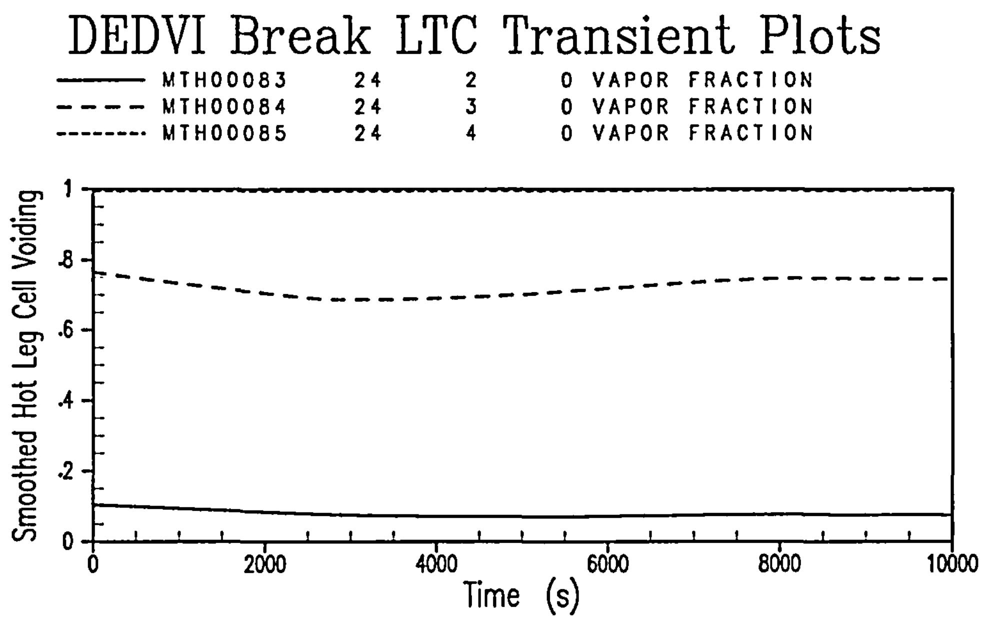


Figure 15.2.7-1-7 Hot Leg Void Fraction

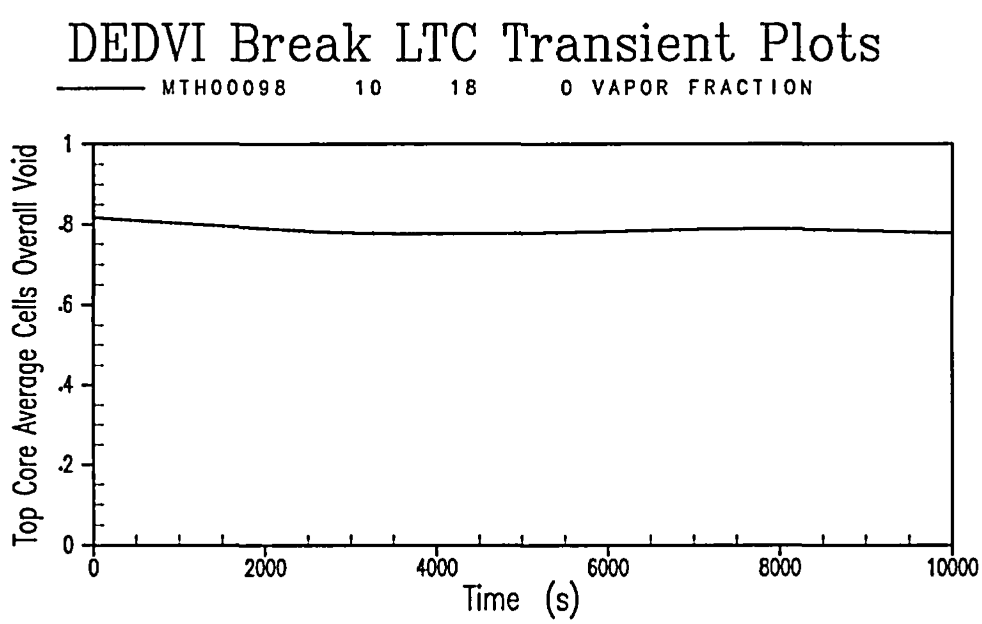


Figure 15.2.7-1-8 Top Core Void Fraction

Attachment A

Summary of Remaining Issue: Provide the following information regarding long-term cooling boron precipitation: (1) the longest time for which the LTC is viable in terms of boron precipitation (i.e., identify the point in time entrainment no longer removes liquid through ADS-4 lines); (2) the shortest reactor operating time which results in sufficient decay heat and steam generation to support boron removal by expulsion of liquid from the ADS-4 valves.

Westinghouse Response:

(1) A simplified, bounding analysis has been performed to establish a conservative operating time for the post LOCA LTC recirculation mode. A bounding model has been constructed which only takes credit for the buoyant effect of the steam as it bubbles up through water in the upper plenum and ADS 4 lines. Other water removal mechanisms that would exist with higher steam flows in annular, churn, slug flow regimes are ignored.

This model incorporates the following conservative assumptions:

1. The core region is assumed to contain no steam voids. Steam voids increase the driving head through the reactor and allow for LTC operation with lower decay heat / longer times.
2. The voiding in the upper plenum and ADS 4 lines are calculated assuming still water pools with no water flow (ie ADS 4 vent quality = 100%). The void in these pools is calculated based on the bubble rise velocity and volumetric flow.
3. The volumetric flow of steam is based on the decay heat levels and the containment pressure and water temperature.
4. The containment pressure is assumed to be 14.7 psia.
5. The containment water is assumed to be subcooled by 20 F.
6. The containment water level is assumed to be at minimum values.

The following equation (equation 9.34, from Wallis, G.B., "One-Dimensional Two-Phase Flow" McGraw-Hill Book Company, 1969) is used to calculate the bubble rise velocity. Note that since the void is a term in the bubble rise velocity equation, the calculation is an iterative one.

$$V = \{ 1.53 * [((st * g * (pl - pg)) / pl^2]^{1/4} \} / (1 - a) \text{, where}$$

V	bubble rise velocity, ft/sec
st	water surface tension, lb/ft
g	gravitational constant, 32.2 ft/sec^2
pl	density of water, lb/ft3
pg	density of steam, lb/ft3
a	void in water

In Table 1, the "Time after shutdown" is the earliest time when water may no longer be removed from the RCS through the ADS 4; at this time steam bubbles in the upper plenum and ADS 4 volumes may not create sufficient void to raise the ADS 4 water level up to the ADS 4 discharge elevation. This time was determined using an iterative method; the time is changed until the head available to inject water from the containment balances the backpressure head caused by the steam / water mixtures in the upper plenum and ADS 4. At later times, there is less steam which leads to less voiding and more backpressure head.

All of these 5 cases in Table 1 use the same inputs / assumptions, except for the containment water level which is varied from the minimum wall-to-wall level (103.3 ft) up to a maximum (110 ft). As expected, higher containment water levels result in longer LTC times. This table uses a conservative minimum safety decay heat level which is based on a condition just after refueling with 33% fresh fuel before the plant has returned to power.

Table 2 shows the LTC limiting time with best estimate decay heat instead of the minimum decay heat used in Table 1. For these cases, the decay heat is based on ANS '79 with no margin added. The LTC times are longer because of the larger amount of decay heat.

Table 1, Case 4 shows that with the water level at 109 ft, the LTC operation is viable for more than 6 years. Even with the minimum wall-to-wall containment water level (103.3 ft), more than 80 days of LTC operation are possible following a LOCA (Case 1). The initial containment minimum flood level is 107.8 ft for a DVI LOCA (in a PXA room) and 109.3 ft for other LOCAs (as shown in Table 3). Operator actions are expected to prevent the containment water level from decreasing to the minimum wall-to-wall level and to maintain / increase that level to more than 109 ft in a week or two.

As a result, with reasonable operator actions the LTC mode of operation is viable for more that 6 years (Table 1, Case 4). These times after shutdown are based on the very conservative assumptions used in this model. Simply by using a more reasonable decay level (ie best estimate decay heat), the LTC times are increased significantly. In addition, most of the cases in Tables 1 and 2 show steam flows (Jg) that are high enough to increase the water carryover above the values calculated in this conservative model.

The operators have IRWST and the containment water level sensors that can measure these levels post accident (refer to DCD Table 7.5-1). There are 4 redundant IRWST level sensors and there are 3 containment water level sensors provided in the plant. The IRWST sensors can measure the full tank volume. The containment water level sensors measure the containment level from the floor under the reactor vessel (elevation 71' 6") up to the maximum containment water level (elevation 110'). DCD Table 7.5-1 shows the range of these sensors from 72' up to 110'. In addition, the operators have the ability to sample the RCS boron concentration in a post LOCA LTC mode. The primary sampling system (PSS) is described in DCD section 9.3.3. The PSS has two HL sample points which can be used to take RCS water samples post accident. The RCS connections are from the bottom of the HL's. The system has an eductor that would be used when the RCS and containment pressures are low (DCD 9.3.3.2.1).

These LTC times are based on no water carryover; some water carryover is needed to limit the boron concentration buildup. The water required to limit the concentration buildup is not large; a quality of 91% limits the boron concentration to 35,000 ppm. Qualities in the 60% range limit the boron concentration to less than 7,500 ppm, which is much less than any concentration limit. The volumetric water flows needed to reduce the quality to these levels are insignificant relative to the steam volumetric flow because of the large density difference at these conditions (density ratio = 26 lb/ft3 / 0.0167 lb/ft3 = 1550). ADS 4 vent qualities in 60% range will not significantly affect the LTC times shown in Tables 1 and 2. Note that as shown in page 2 of this response, both WCOBRA-TRAC and simplified models predict that the ADS 4 exit qualities will actually be very low (< 10%) in these longer times.

(2) The decay heat from the initial core has been analyzed for short irradiation times. For such a situation, the long-term decay heat decreases more rapidly than for an equilibrium core as shown in Figure 1.

The probability of having a LOCA early in the first core life is very low. The following shows the probability of having a small or medium LOCA (with an assumed LOCA frequency of 1 E -3 / yr) during the first year as well as during the whole plant life (60 years):

Irradiation Time	Prob. in 1st Year	Avg .Prob. in Plant Life
1 day	2.7 E -6	4.6 E -8 / yr
7 days	1.9 E -5	3.2 E -7 / yr
14 days	3.8 E -5	6.4 E -7 / yr
30 days	8.2 E -5	1.4 E -6 / yr

Considering a 1 week irradiation time, the probability of having a LOCA during this time with the first core is very low. Therefore, it is reasonable to analyze such a low probability event using best estimate assumptions. The best estimate containment water level is greater than 107 ft for at least 3 days for a DVI LOCA with no injection of the BAT or the RNS (Table 3, Case 3). For a non-DVI LOCA, the best estimate level is greater than 108 ft for at least three days (Table 3, Case 4). Three days is more than enough time for the operators to add water to the containment to maintain and increase the water level. A minimum containment water level of 107 ft is used in this analysis.

Table 1 (Case 2) and 2 (Case 7) show that the minimum decay heat required with a containment level of 107 ft is 433.2 BTU/sec. From Figure 1, the time when this decay heat would be reached for a first core irradiated for 7 days would be more than 30 days after shutdown. This will provide the operators more than enough time to take actions to maintain and increase the containment water level. Higher water levels provide additional LTC times for a first core irritated for 7 days.

Cont Level (ft)	DH (BTU/sec)	LTC Time (Days)
107	433.2	33
108	294.8	45
109	186.9	67

110	103.4	107

In conclusion, this conservative, bounding analysis shows that the AP1000 provides viable post LOCA LTC operation for a very long time (> 6 years) such that excessive boron concentration buildup will not occur.

Table 1 – AP1000 Limiting Long-Term Cooling Conditions With Minimum Decay Heat

	Case 1	Case 2	Case 3	Case 4	Case 5
Time after shutdown (days)	88.3	823.2	1,313.1	2,497.6	7,069.2
, (years)	0.2	2.3	3.6	6.8	19.4
Decay heat basis	Min	Min	Min	Min	Min
Decay heat, BTU/sec	1776.7	433.2	294.8	186.9	103.4
Cont pres (psia)	14.7	14.7	14.7	14.7	14.7
Cont water temp (F)	192.0	192.0	192.0	192.0	192.0
Water subcooling (F)	20.0	20.0	20.0	20.0	20.0
Cont water level, ft	103.3	107.0	108.0	109.0	110.0
ADS 4 vent quality	100%	100%	100%	100%	100%
Core steam voiding	0%	0%	0%	0%	0%
Flows, Total (lb/sec)	1.794	0.437	0.298	0.189	0.104
, Steam (lb/sec)	1.794	0.437	0.298	0.189	0.104
, Water (lb/sec)	0.000	0.000	0.000	0.000	0.000
Void, upper core	0.0%	0.0%	0.0%	0.0%	0.0%
, upper plenum	19.7%	5.3%	3.6%	2.3%	1.2%
, ADS 4	79.2%	47.3%	37.5%	27.4%	17.2%
Bubble rise, core (ft/sec)	na	na	na	na	na
, upper plenum (ft/sec)	2.31	1.95	1.92	1.89	1.87
, ADS 4 pipe (ft/sec)	8.93	3.53	3.00	2.57	2.25
Jg (gas vel), m/sec	6.27	1.48	1.00	0.63	0.34
Jl (liq vel), m/sec	0.00	0.00	0.00	0.00	0.00
Densities (lb/ft3)					
Rrecirc / downcomer	60.30	60.30	60.30	60.30	60.30
Core	59.81	59.81	59.81	59.81	59.81
Upper plenum	49.74	56.47	57.37	58.07	58.63
ADS 4 pipe	13.22	31.92	37.64	43.50	49.48
Pressures (psia)					
Containment	14.70	14.70	14.70	14.70	14.70
Bottom fuel	23.76	25.31	25.73	26.15	26.56
Top fuel	17.94	19.49	19.91	20.33	20.75
Top HL	15.57	16.80	17.17	17.56	17.95
ADS 4 discharge	14.70	14.70	14.70	14.70	14.70

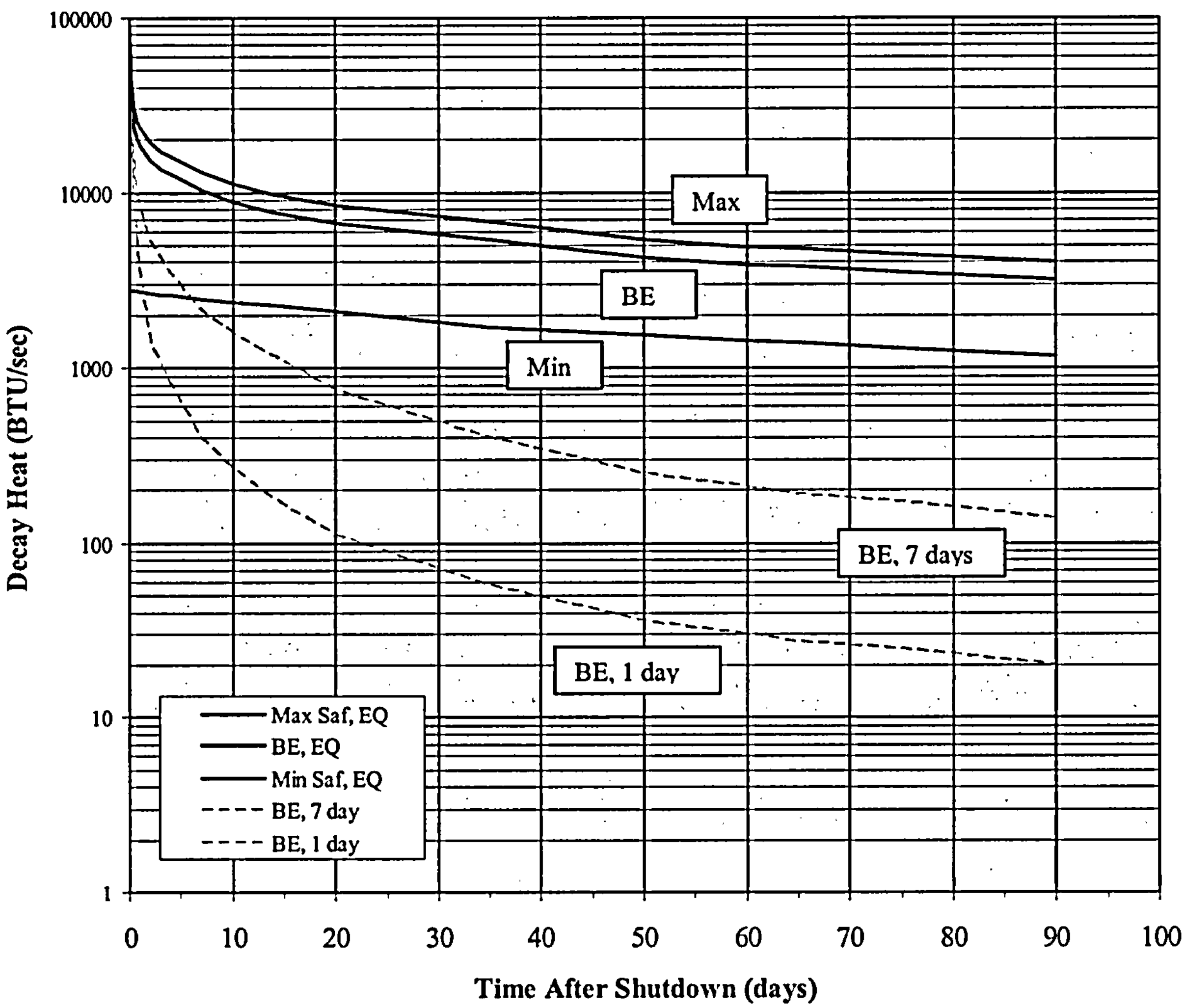
Table 2 – AP1000 Limiting Long-Term Cooling Conditions With Best Estimate Decay Heat

	Case 6	Case 7	Case 8	Case 9	Case 10
Time after shutdown (days)	425.3	1,465.9	2,585.3	5,566.5	12,146.1
, (years)	1.2	4.0	7.1	15.3	33.3
Decay heat basis	BE	BE	BE	BE	BE
Decay heat, BTU/sec	1231.8	433.2	293.0	186.4	103.4
Cont pres (psia)	14.7	14.7	14.7	14.7	14.7
Cont water temp (F)	192.0	192.0	192.0	192.0	192.0
Water subcooling (F)	20.0	20.0	20.0	20.0	20.0
Cont water level, ft	104.2	107.0	108.0	109.0	110.0
ADS 4 vent quality	100%	100%	100%	100%	100%
Core steam voiding	0%	0%	0%	0%	0%
Flows, Total (lb/sec)	1.244	0.437	0.296	0.188	0.104
, Steam (lb/sec)	1.244	0.437	0.296	0.188	0.104
, Water (lb/sec)	0.000	0.000	0.000	0.000	0.000
Void, upper core	0.0%	0.0%	0.0%	0.0%	0.0%
, upper plenum	14.3%	5.3%	3.6%	2.2%	1.2%
, ADS 4	72.3%	47.3%	37.5%	27.4%	17.2%
Bubble rise, core (ft/sec)	na	na	na	na	na
, upper plenum (ft/sec)	2.16	1.95	1.92	1.89	1.87
, ADS 4 pipe (ft/sec)	6.73	3.53	2.98	2.56	2.25
Jg (gas vel), m/sec	4.32	1.48	0.99	0.62	0.34
Jl (liq vel), m/sec	0.00	0.00	0.00	0.00	0.00
Densities (lb/ft3)					
Rrecirc / downcomer	60.30	60.30	60.30	60.30	60.30
Core	59.81	59.81	59.81	59.81	59.81
Upper plenum	52.07	56.47	57.37	58.08	58.63
ADS 4 pipe	17.27	31.92	37.64	43.50	49.48
Pressures (psia)					
Containment	14.70	14.70	14.70	14.70	14.70
Bottom fuel	24.14	25.31	25.73	26.15	26.56
Top fuel	18.32	19.49	19.91	20.33	20.75
Top HL	15.83	16.80	17.17	17.56	17.95
ADS 4 discharge	14.70	14.70	14.70	14.70	14.70

Table 3 – AP1000 Containment Flood Elevations

	Case 1	Case 2	Case 3	Case 4	Case 5
Level Basis	Min	Min	BE	BE	Max
LOCA	DVI	Non DVI	DVI	Non DVI	Non DVI
Cont Flood Vol.	Max	Max	BE	BE	Min
Water Supplies					
RCS/PXS	Min	Min	BE	BE	Max
BAT	none	Min	none	BE	Max
RNS	none	none	none	BE	Max
Containment Water Level					
Initial recirc	107.8	109.3	108.2	110.0	110.2 ft
1 days	107.4	108.8	107.9	109.5	109.7 ft
3 days	106.2	107.9	107.1	108.6	108.8 ft
7 days	104.4	106.2	105.1	107.4	107.9 ft
14 days	103.3	104.2	103.9	106.9	107.9 ft
30 days	103.3	104.2	103.9	106.9	107.9 ft

Figure 1 – AP1000 Decay Heat Values



Shutdown (days)	Equil Core 6 year Max Saf	Equil Core 6 year BE	Equil Core 6 year Min Saf	First Core 1 day BE	First Core 7 day BE	
0.417	30148	83435	65967	2806	50562	BTU/sec
1	23433	18527	2748	3367	8204	BTU/sec
7	13012	10288	2474	390	2106	BTU/sec
14	9807	7754	2283	179	1177	BTU/sec
35	6807	5382	1695	57	402	BTU/sec
90	4055	3206	1174	20	139	BTU/sec

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 21.5-2 Item APEX Scaling

Original RAI Number(s): None

NRC Comments from 12/17/03 Status meeting:

APEX-AP1000 Scaling Report Questions

1. Clarify the units listed in Tables 7, 8, and 9 of Reference 1. Verify that the unit listed $(ft/gpm^2)^2$ should be (ft/gpm^2). In addition, please correct the min and max headings. They are reversed. In addition, fix the typos ("MT" should be "CMT" in Table 8, and "DS-4" should be "ADS-4" in Table 9. These typos are also in the Table of Contents, page ix.)

2. Please clarify the differences between the resistances in Tables 7, 8, and 9 of Reference 1 and Table 3-14 of Reference 2. The values do not always agree. For example, in Table 3-14 the line resistance for CMT-1 is listed as 5.523x10-2 ft/gpm^2. In Table 8, there is no distinction between CMT-1 and CMT-2. The Table 3-14 value does not agree with any of the Table 8 CMT values. For both AP1000 and APEX-AP1000, specify the line resistances for CMT-1 and CMT-2.

3. Please clarify the correct value of RCMT-DVI for the AP1000 plant. In Attachment 2 of Reference 3, Westinghouse reported this to be 261 for AP1000. This does not agree with the equivalent value derived from Table 8 of Reference 1.

4. What is the total (free) volume within the AP1000 reactor vessel. Verify that 0.551 m^3 represents the free volume in the APEX-AP1000 vessel.

5. What is the (free) volume within the AP1000 and APEX-AP1000 reactor vessels below the bottom of the active fuel/heaters?

6. Table 12 of Reference 1 lists the ADS-4 two trains flow area as 864.51 cm^2 for the AP1000 plant. This does not agree with the AP1000 value in Reference 3, Attachment 4 value for the total ADS-4 outlet flow area. The Reference 1 value does appear to be close to the AP600 value. Please clarify the correct values in References 1 and 3 and show that the ADS-4 flow area was scaled using the correct AP1000 value.

7. In the AP1000 plant design, the pipe flow area at the ADS-4/hot leg nozzle is greater than the flow area at the ADS-4 branch line. (See Table 9 of Reference 1.) In the APEX design, it is the opposite, with the area at the branch line greater than at the ADS-4/hot leg nozzle. Please explain why this represents appropriate scaling.

8. In the APEX-AP600 scaling report, the ADS-4 was listed in Table 7-20 as having three separate sections. Section 1 had an inside diameter of 1.61 inches, Section 2 had an ID of 1.689 inches, and Section 3 had an ID of 2.06 inches. Define section(s) and line diameters for APEX-AP1000, as only two values are listed in Table 9 of Reference 1, and

in Reference 2, the ADS-4 is listed as being "2 in, sch. 40" in Table 3-13, while being "2.5 in, sch 40" in the text on page 3-4 of that same reference. Please clarify the ADS-4 line size(s) in the APEX-AP1000 facility. In addition, clarify if the scaling performed in Reference 1 corresponds to "no nozzle," "50% nozzle," or "100% nozzle" line resistance with regard to Table 3-14 of Reference 2.

9. For APEX tests DBA-02 and DBA-03, provide a tabular listing of total RCS mass versus time, pressurizer mass versus time, and pressure versus time. The total RCS mass should represent the mass within the vessel, the hot and cold legs, pumps, and primary side of the steam generators. For the pressurizer mass, include mass held within the surge line. For system pressure, use the upper head pressure.

   Likewise, provide the total RCS mass versus time, pressurizer mass versus time, and pressure versus time as described above for NOTRUMP simulations of AP1000 double-ended DVI line breaks that correspond to tests DBA-02 and DBA-03. (A DVI line break with single ADS-4 valve failures on the pressurizer side, and another with the failure assumed on the non-pressurizer side.

10. Provide or verify the final design values for the AP1000 and APEX as configured for AP1000 testing. Figure 2-2 of Reference 2 shows the pressurizer attached to Loop 2. So, let loops "1" or "A" designate the non-pressurizer loop while loops "2" or "B" represent the pressurizer side. Line resistances should represent $R = K/A^2$ such that the single phase pressure drop across the line is given by $\Delta P = R * 0.5 * \rho V^2$.

Parameter	AP1000	APEX
R_{cmt1} ; CMT-1 resistance from CMT to vessel	2.03e-05 ft/gpm^2	4.68e-02 ft/gpm^2
R_{cmt2} ; CMT-2 resistance from CMT to vessel	2.03e-05 ft/gpm^2	4.68e-02 ft/gpm^2
$(L/A)_{cmt1}$; Inertial length from CMT-1 to DVI nozzle		
$(L/A)_{cmt2}$; Inertial length from CMT-2 to DVI nozzle		
$\Delta Z_{cmt\text{-}dvi}$; Elevation change from bottom of CMT to DVI centerline	2.309 m	0.743 m
$\Delta Z_{cmt\text{-}boc}$; Elevation change from bottom of CMT to bottom of active core		
V_{RPV} ; RPV volume		
V_{UP} ; Upper plenum volume		
V_{LP} ; RPV volume below active core		
$\Delta Z_{prz\text{-}hl}$; Elevation change from bottom of pressurizer to HL centerline		
$\Delta Z_{hl\text{-}dvi}$; Elevation difference between hot leg centerline and DVI line centerline	12.42 ft	3.104 ft
$\Delta Z_{dvi\text{-}boc}$; Elevation difference between DVI line centerline and bottom of active core		

Parameter	AP1000	APEX
R_{SL} ; Pressurizer surge line resistance		
L_{SL} ; Surge line total length		
R_{dvi1} ; Line resistance from IRWST to vessel through DVI line 1		
R_{dvi2} ; Line resistance from IRWST to vessel through DVI line 2		
L_{dvi1} ; Line length from IRWST to vessel through DVI line 1		
L_{dvi2} ; Line length from IRWST to vessel through DVI line 2		
$\Delta Z_{irwst-dvi}$; Elevation change from bottom of IRWST to DVI line centerline		
ΔZ_{irwst} ; IRWST minimum level (referenced from bottom of IRWST tank)	343 in	91.9 in
R_{ads1} ; ADS-4-1 line resistance (single valve failure)	4.89e-07 ft/gpm^2	1.13e-03 ft/gpm^2
R_{ads1} ; ADS-4-1 line resistance (no failure)	1.36e-07 ft/gpm^2	3.14e-04 ft/gpm^2
R_{ads2} ; ADS-4-2 line resistance (single valve failure)	4.98e-07 ft/gpm^2	1.15e-03 ft/gpm^2
R_{ads2} ; ADS-4-2 line resistance (no failure)	1.26e-07 ft/gpm^2	2.90e-04 ft/gpm^2
$\Delta z_{ads-boc}$; Elevation difference between ADS-4 discharge to containment and bottom of active core		
L_{ads1} ; Length of Loop 1 ADS-4 piping		
L_{ads2} ; Length of Loop 2 ADS-4 piping		
$\Delta Z_{sump-boc}$; Elevation difference between maximum sump level (determined by curb height) and bottom of active core		

11. Editorial comment: Should ρ_{liter} be ρ_{liquid} in Equation (168)?

12. Editorial comment: There is a problem with the font in Equations (198), (199), (200), (214), (229), (232), and (235). The "LE" should appear as a less than or equal to sign.

13. For tests DBA-02 and DBA-03, provide the upper plenum mass as a function of time and also the system pressure (pressurizer or RPV upper head).

References:

1. OSU-APEX-03001, Scaling Assessment for the Design of the OSU APEX-1000 Test Facility.
2. OSU-APEX-03002, OSU APEX-1000 Test Facility Description Report.
3. DCP/NRC1484, September 12, 2001.

Westinghouse Response:

1. *Clarify the units listed in Tables 7, 8, and 9 of Reference 1. Verify that the unit listed $(ft/gpm^2)^2$ should be (ft/gpm^2). In addition, please correct the min and max headings. They are reversed. In addition, fix the typos ("MT" should be "CMT" in Table 8, and "DS-4" should be "ADS-4" in Table 9. These typos are also in the Table of Contents, page ix.)*

   The unit listed should be ft/gpm^2 in Tables 7, 8, and 9 of Reference 1. The typos listed will be corrected in a subsequent revision of Reference 1.

2. *Please clarify the differences between the resistances in Tables 7, 8, and 9 of Reference 1 and Table 3-14 of Reference 2. The values do not always agree. For example, in Table 3-14 the line resistance for CMT-1 is listed as 5.523x10-2 ft/gpm^2. In Table 8, there is no distinction between CMT-1 and CMT-2. The Table 3-14 value does not agree with any of the Table 8 CMT values. For both AP1000 and APEX-AP1000, specify the line resistances for CMT-1 and CMT-2.*

   We have reviewed the AP1000 values Tables 7, 8, and 9 of Reference 1 and found them to be correct, with the following minor comments:
   - Table 7: In the sections for "IRWST to Sump Tee" and "IRWST Drain to Containment" different I.D.'s are shown for Line A and Line B. This is misleading; both Line A and B have both of these line sizes. Thus for all four of these I.D. entries it would have been more correct to show "10.02 and 7.981".
   - Table 7: In the section "IRWST Drain to Containment" there is a typo: the nominal resistance shows 3.36E-06 and should show 3.26E-06 (i.e., it should be identical to Line A).

   We have not established separate resistances for CMT Line B, but it will have lower resistance than Line A; thus Line A resistance is conservatively used.

   The difference between the values reported in the Facility Description Report and the Scaling Report is the Scaling Report lists the IDEAL scaled values. These values are determined by scaling the appropriate AP1000 Line Resistances. The line resistance scaling ratio for single-phase fluid under non-choked flow conditions is given as:

$$R_R = 48^2 = 2304$$

Multiplying the AP1000 Line Resistance Values by 2304 results in the values listed in the scaling report. Because the AP1000 line resistance values were broken into sections, the Scaling Report follows this convention.

The values reported in the Facility Description Report were determined by performing a Flow Test for the entire line. Please note that the values listed in the Facility Description Report for ADS-4 are for the 2 in sch 40 ADS-4 piping configuration. The 2 ½ inch sch 40 ADS-4 configuration (w/ venturis) are not listed but are provided below in the response to item 10.

The line resistance values as determined by the flow test include the total line and are not broken into sections as listed in the scaling report. The appropriate values from the scaling report are added to arrive at an ideal value for the total line resistance.

3. *Please clarify the correct value of RCMT-DVI for the AP1000 plant. In Attachment 2 of Reference 3, Westinghouse reported this to be 261 for AP1000. This does not agree with the equivalent value derived from Table 8 of Reference 1.*

   Although this resistance value changed slightly between the issue of Reference 3 and Reference 1, the change was minor (less than 1%). Using Reference 1 Table 7 nominal values, this resistance will be the sum of "CMT to ACC Tee" plus "ACC to DVI" - thus 1.62E-5 + 4.08E-6 ft/gpm^2, which is 2.03E-5 ft/gpm^2. Converting to ft^{-4} with a factor of 1.29E7 gives 261.5 ft^{-4}.

4. *What is the total (free) volume within the AP1000 reactor vessel. Verify that 0.551 m^3 represents the free volume in the APEX-AP1000 vessel.*

   The total free volume within the AP1000 reactor vessel is

   $V_{vessel\text{-}AP1000} = 3559.75\ ft^3\ (100.786\ m^3)$

   The total free volume within the APEX1000 reactor vessel is

   $V_{vessel\text{-}APEX1000} = 20.004\ ft^3\ (0.5504\ m^3)$

5. *What is the (free) volume within the AP1000 and APEX-AP1000 reactor vessels below the bottom of the active fuel/heaters?*

   The free volume in the AP1000 reactor vessel below the fuel is

   $V_{LP\text{-}AP1000} = 440.31\ ft^3$

   The free volume below the heater rods for the APEX1000 reactor vessel is

   $V_{LP\text{-}APEX1000} = 2.983\ ft^3$

6. *Table 12 of Reference 1 lists the ADS-4 two trains flow area as 864.51 cm^2 for the AP1000 plant. This does not agree with the AP1000 value in Reference 3, Attachment 4 value for the total ADS-4 outlet flow area. The Reference 1 value does appear to be close to the AP600 value. Please clarify the correct values in References 1 and 3 and show that the ADS-4 flow area was scaled using the correct AP1000 value.*

   The value provided in Table 12 of Reference 1 is correct; two valves, each with an area of about 67 in^2 are provided. The appropriate value was used for AP1000 scaling.

7. *In the AP1000 plant design, the pipe flow area at the ADS-4/hot leg nozzle is greater than the flow area at the ADS-4 branch line. (See Table 9 of Reference 1.) In the APEX design, it is the opposite, with the area at the branch line greater than at the ADS-4/hot leg nozzle. Please explain why this represents appropriate scaling.*

   The flow area at the ADS-4/Hot Leg nozzle is greater than the downstream flow area of the ADS-4 branch pipe for the AP1000 plant whereas in the APEX-1000 test facility, the downstream flow area is slightly larger than the flow area at the ADS-4/Hot Leg nozzle. This is due to the fact that the APEX test facility was designed to simulate two ADS-4 branch pipes with a single pipe. Different flow nozzles are interchanged in the ADS-4 line of the APEX test facility to simulate one or two flow path operation.

   For critical flow conditions through the ADS-4 flow paths, the flow area of the piping is not important as the ADS-4 nozzle is most limiting. For non-critical flow conditions through ADS-4 flow paths, the ADS-4 pipe area impacts acceleration pressure drop. The impact of the ADS-4 flow area on acceleration pressure drop downstream of the core exit (i.e. ADS-4 flow path) is best represented by examining the area of the hot leg relative to the ADS-4 branch pipe. The flow from the hot leg is accelerated toward the ADS-4 discharge. Comparison of the hot leg flow area relative to ADS-4 branch line flow area indicates that APEX-1000 is well scaled to AP1000 for conditions simulating two branch flow path operation. The ratio between the test facility and plant of the area of the hot leg relative to the ADS-4 pipe flow area (A_{hotleg}/A_{ads4}) is about 1.0. For simulating single branch flow path operation (i.e. corresponding to single failure of one ADS-4 valve), APEX is not as well scaled as the ratio is about ½. Therefore, the ADS-4 acceleration pressure drop is expected to be well represented for simulations involving no failures in the ADS-4 branch lines but somewhat understated for simulations of a single failure in one of the ADS-4 branch lines. The overall impact of this on the ADS-4 pressure drop is tempered for simulation of design basis accidents as it only affects one of the two trains of the ADS-4 and the form and friction pressure losses are appropriately scaled for both trains.

8. *In the APEX-AP600 scaling report, the ADS-4 was listed in Table 7-20 as having three separate sections. Section 1 had an inside diameter of 1.61 inches, Section 2 had an ID of 1.689 inches, and Section 3 had an ID of 2.06 inches. Define section(s) and line diameters for APEX-AP1000, as only two values are listed in Table 9 of Reference 1, and in Reference 2, the ADS-4 is listed as being "2 in, sch. 40" in Table 3-13, while being "2.5 in, sch 40" in the text on page 3-4 of that same reference. Please clarify the ADS-4 line size(s) in the APEX-AP1000 facility. In addition, clarify if the scaling performed in*

Reference 1 corresponds to "no nozzle," "50% nozzle," or "100% nozzle" line resistance with regard to Table 3-14 of Reference 2.

For APEX1000 the ADS-4 lines were changed to 2-inch, schedule 40 pipe. After the first DOE test, DBA-01, it was determined that the unchoked flow in the ADS-4 pipe was less than the ideally scaled flow. The ADS-4 lines were replaced with 2.5-inch, schedule 40 pipe and the ADS-4 nozzles that represent the valves were changed to venturis. This arrangement was used for all subsequent tests in APEX1000. The piping is shown in the as-built drawings below. There is a 2-inch nozzle off the hot leg that expands to 2.5-inch pipe that extends to the isolation valve which is a 3-inch ball valve. After the valve, 2.5-inch pipe extends to the venturi. After the venturi, 2.5-inch pipe extends to a 3.5-inch diameter expansion section that is 4-inches long at the ADS-4 separator. The two ADS-4 lines use the same size piping.

Table 9 of Reference 1 shows ideal ADS-4/Hot Leg nozzle and ADS-4 branch lines sizes based upon scaling analysis for APEX-1000. The actual ADS-4 line sizes in the APEX-1000 test facility are shown in the drawings below.

The ADS-4 ideal line resistances based upon scaling analysis are shown for "50% nozzle" and "100% nozzle" configurations in APEX-1000 in Table 9 of Reference 1. The actual APEX-1000 line resistances obtained via flow test are provided below in the response to item 10 for each line, with a 50% venturi (single failure) and with a 100% venturi (no failure).

a,c

a,c

9. *For APEX tests DBA-02 and DBA-03, provide a tabular listing of total RCS mass versus time, pressurizer mass versus time, and pressure versus time. The total RCS mass should represent the mass within the vessel, the hot and cold legs, pumps, and primary side of the steam generators. For the pressurizer mass, include mass held within the surge line. For system pressure, use the upper head pressure.*

   Likewise, provide the total RCS mass versus time, pressurizer mass versus time, and pressure versus time as described above for NOTRUMP simulations of AP1000 double-ended DVI line breaks that correspond to tests DBA-02 and DBA-03. (A DVI line break with single ADS-4 valve failures on the pressurizer side, and another with the failure assumed on the non-pressurizer side.

   The plots of RCS mass (without the upper head and pressurizer), pressurizer mass and upper head pressure versus time for Tests DBA-02 and DBA-03 are shown in Figures 1-6. The upper head mass has been removed from both RCS mass plots due to questions related to the upper head response in test DBA-03. The corresponding AP1000 plant simulations comparable to tests DBA-02 and DBA-03 are included in Figures 7-12. Both of the AP1000 simulations were performed with 25.0 psi containment backpressure. The tabular form of the data is included in the attached CD.

a,c

Figure 1: RCS Mass for DBA-02

a,c

Figure 2: Pressurizer Mass for DBA-02

a.c

[ ]

Figure 3: Upper Head Pressure for DBA-02

a,c

Figure 4: RCS Mass for DBA-03

a,c

Figure 5: Pressurizer Mass for DBA-03

a,c

Figure 6: Upper Head Pressure for DBA-03

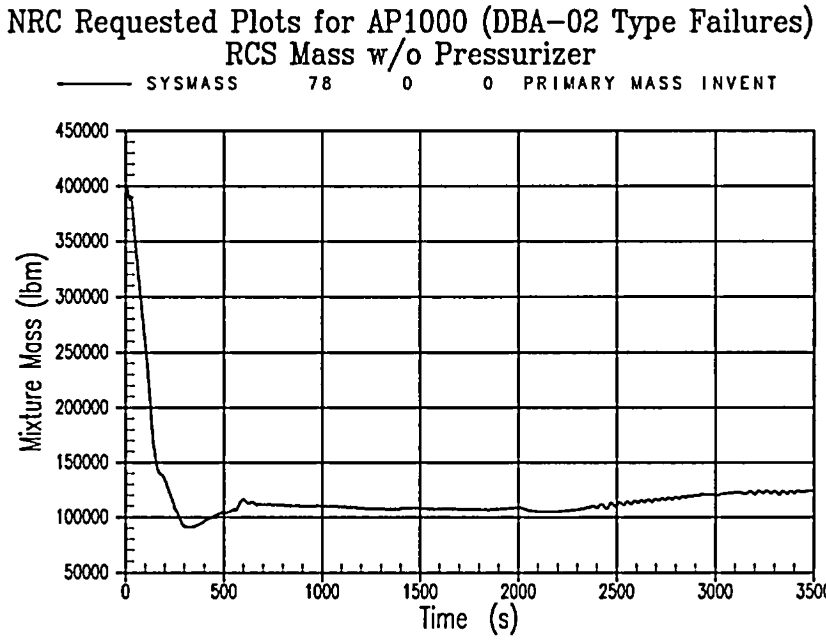


Figure 7: RCS Mass for AP1000 (DBA-02 type failures), 25 psi Containment

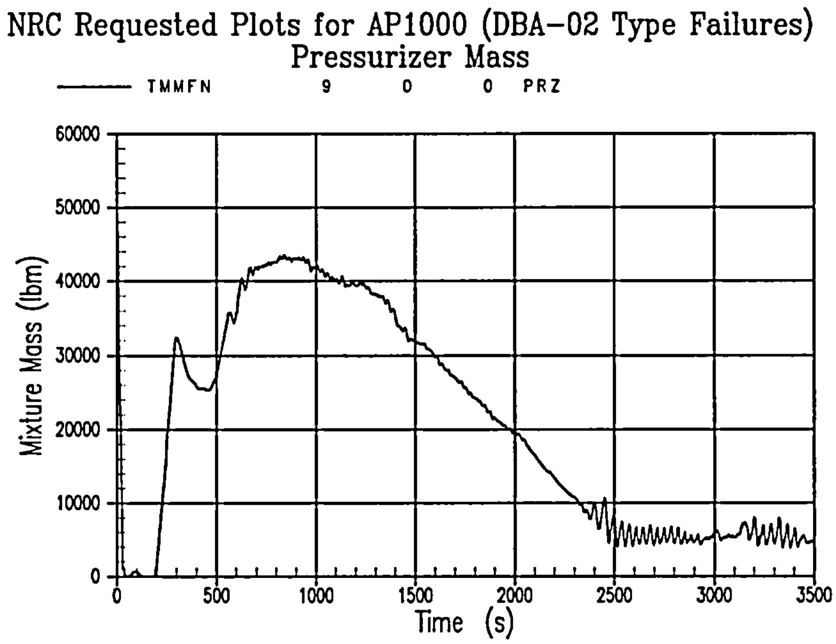


Figure 8: Pressurizer Mass for AP1000 (DBA-02 type failures), 25 psi Containment

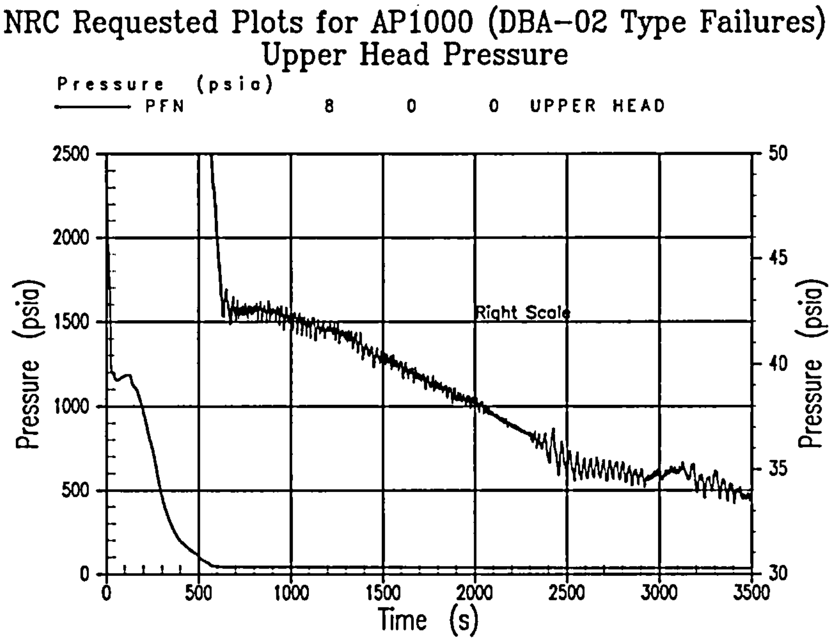


Figure 9: Upper Head Pressure for AP1000 (DBA-02 type failures), 25 psi Containment

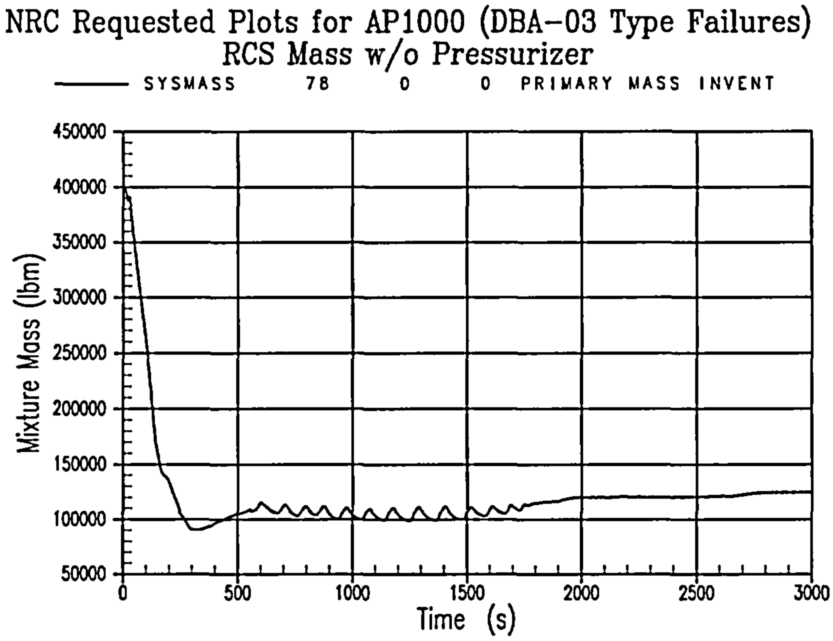


Figure 10: RCS Mass for AP1000 (DBA-03 type failures), 25 psi Containment

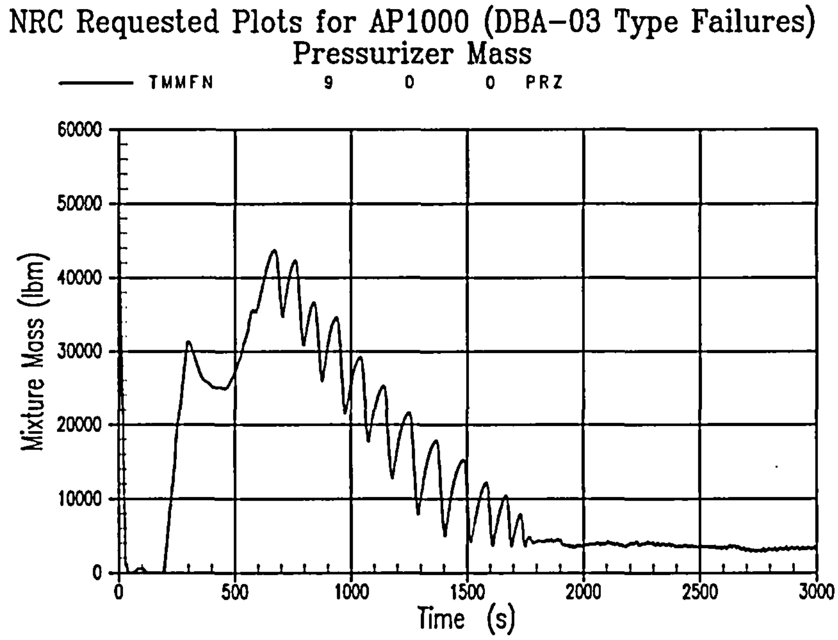


Figure 11: Pressurizer Mass for AP1000 (DBA-03 type failures), 25 psi Containment

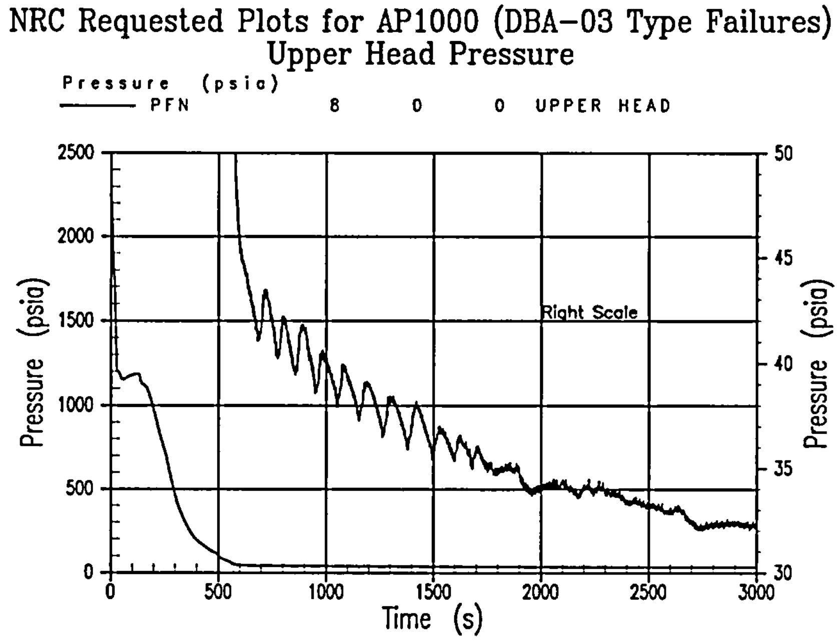


Figure 12: Upper Head Pressure for AP1000 (DBA-03 type failures), 25 psi Containment

10. *Provide or verify the final design values for the AP1000 and APEX as configured for AP1000 testing. Figure 2-2 of Reference 2 shows the pressurizer attached to Loop 2. So, let loops "1" or "A" designate the non-pressurizer loop while loops "2" or "B" represent the pressurizer side. Line resistances should represent $R = K/A^2$ such that the single phase pressure drop across the line is given by $\Delta P = R * 0.5 * \rho V^2$.*

Requested parameters are provided in the following table.

a,c

Parameter	AP1000	APEX
R_{cmt1} ; CMT-1 resistance from CMT to vessel	3.547e-05 ft/gpm^2	
R_{cmt2} ; CMT-2 resistance from CMT to vessel	3.439e-05 ft/gpm^2	
$(L/A)_{cmt1}$; Inertial length from CMT-1 to DVI nozzle	257.9 ft-1	
$(L/A)_{cmt2}$; Inertial length from CMT-2 to DVI nozzle	258.6 ft-1	
$\Delta Z_{cmt\text{-}dvi}$; Elevation change from bottom of CMT to DVI centerline	2.309 m	
$\Delta Z_{cmt\text{-}boc}$; Elevation change from bottom of CMT to bottom of active core	25.62 ft	
V_{RPV} ; RPV volume	3559.75 ft3	
V_{UP} ; Upper plenum volume	691.91 ft3	
V_{LP} ; RPV volume below active core	440.31 ft3	
$\Delta Z_{prz\text{-}hl}$; Elevation change from bottom of pressurizer to HL centerline	19.23 ft	
$\Delta Z_{hl\text{-}dvi}$; Elevation difference between hot leg centerline and DVI line centerline	1.67 ft	
$\Delta Z_{dvi\text{-}boc}$; Elevation difference between DVI line centerline and bottom of active core	26.36 ft	
R_{SL} ; Pressurizer surge line resistance	1.796e-7 ft/gpm^2	
L_{SL} ; Surge line total length	88.87 ft	
R_{dvi1} ; Line resistance from IRWST to vessel through DVI line 1	9.206e-6 ft/gpm^2	
R_{dvi2} ; Line resistance from IRWST to vessel through DVI line 2	1.035e-5 ft/gpm^2	
L_{dvi1} ; Line length from IRWST to vessel through DVI line 1	85.19 ft	
L_{dvi2} ; Line length from IRWST to vessel through DVI line 2	111.50 ft	
$\Delta Z_{irwst\text{-}dvi}$; Elevation change from bottom of IRWST to DVI line centerline	3.43 ft	
ΔZ_{irwst} ; IRWST minimum level (referenced from bottom of IRWST tank)	343 in	

a,c

Parameter	AP1000	APEX
R_{ads1} ; ADS-4-1 line resistance (single valve failure)	6.114e-07 ft/gpm^2	
R_{ads1} ; ADS-4-1 line resistance (no failure)	1.705e-07 ft/gpm^2	
R_{ads2} ; ADS-4-2 line resistance (single valve failure)	6.221e-07 ft/gpm^2	
R_{ads2} ; ADS-4-2 line resistance (no failure)	1.574e-07 ft/gpm^2	
$\Delta z_{ads\text{-}boc}$; Elevation difference between ADS-4 discharge to containment and bottom of active core	31.36 ft	
L_{ads1} ; Length of Loop 1 ADS-4 piping	14.86 ft	
L_{ads2} ; Length of Loop 2 ADS-4 piping	18.90 ft	
$\Delta Z_{sump\text{-}boc}$; Elevation difference between maximum sump level (determined by curb height) and bottom of active core	29.53 ft	

11. *Editorial comment: Should ρ_{liter} be ρ_{liquid} in Equation (168)?*

    Yes, this will be corrected in a subsequent revision.

12. *Editorial comment: There is a problem with the font in Equations (198), (199), (200), (214), (229), (232), and (235). The "LE" should appear as a less than or equal to sign.*

    Yes, this will be corrected in a subsequent revision.

13. *For tests DBA-02 and DBA-03, provide the upper plenum mass as a function of time and also the system pressure (pressurizer or RPV upper head).*

    Note that the system pressure for these tests is shown in the response for item 9. The upper plenum mass is shown in Figures 13 and 14.

a,c

Figure 13: Upper Plenum Mass for DBA-02

a,c

Figure 14: Upper Plenum Mass for DBA-03

Design Control Document (DCD) Revision:

None

PRA Revision:

None