Westinghouse

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U.S. Nuclear Regulatory Commission
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Your ref. Docket No. 52-006
Our ref DCP/NRC1555

March 17, 2003

SUBJECT: Transmittal of Revised Westinghouse Proprietary and Non-Proprietary Responses to U.S. Nuclear Regulatory Commission Requests for Additional for the AP1000 Application for Design Certification

This letter transmits the revised Westinghouse responses to NRC Requests for Additional Information (RAI) regarding our application for Design Certification of the AP1000 standard plant. The list of RAI responses that are transmitted with this letter is provided in Attachment 1. Attachments 2 and 3 to this letter provide the proprietary and non-proprietary responses to the NRC RAI.

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. Attachment 3 contains no proprietary information.

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

Docket No. 52-006
DCP/NRC1555

March 17, 2003

Correspondence with respect to the application for withholding should reference AW-03-1611, and should be addressed to Hank A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania, 15230-0355.

Please contact me if you have questions regarding this submittal.

Very truly yours,

M. M. Corletti

M. M. Corletti
Passive Plant Projects & Development
AP600 & AP1000 Projects

/Enclosure

1. Westinghouse Electric Company Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit AW-03-1611

/Attachments

1. Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1555"
2. Westinghouse Revised Proprietary Response to US Nuclear Regulatory Commission Requests for Additional Information dated March 17, 2003
3. Westinghouse Revised Non-Proprietary Responses to US Nuclear Regulatory Commission Requests for Additional Information dated March 17, 2003

DCP/NRC1555

March 17, 2003

Enclosure 1

Westinghouse Electric Company
Application for Withholding, Affidavit, Copyright Notice, Proprietary Information Notice

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

March 17, 2003

AW-03-1611

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 Revised Proprietary Class 2 and Non-Proprietary Class 3 versions of Document: "AP1000 Design Certification Review – Responses to Requests for Additional Information"

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-03-1611 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-03-1611 and should be addressed to the undersigned.

Very truly yours,

M. M. Corletti

M. M. Corletti
Passive Plant Projects & Development
AP600 / AP1000 Projects

/Enclosures

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AW-03-1611

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
Westinghouse Electric Company, LLC

Sworn to and subscribed
before me this 17th day
of March, 2003

Notary Public

LORRAINE M. PIPLICA
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

(1) I am Manager, Passive Plant Projects & Development, in the Nuclear Power Plants Business Unit, 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.

The proprietary information sought to be withheld in this submittal is that which is appropriately marked in Attachment 1 as Proprietary Class 2 in the Westinghouse document DCP/NRC1555 for submittal to the Commission: (1) "AP1000 Design Certification Review – Revised Response to Requests for Additional Information."

This information is being transmitted by Westinghouse's letter and Application for Withholding Proprietary Information from Public Disclosure, being transmitted by Westinghouse Electric Company (W letter AW-03-1611) and to the Document Control Desk, Attention: John Segala, DIPM/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 Certification 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.

DCP/NRC1555

Docket 52-006
March 17, 2003

Copyright Notice

Docket 52-006
March 17, 2003

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

Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1555"

Attachment 1

Table 1 "List of Westinghouse's Revised Responses to RAIs as Transmitted in DCP/NRC1555"
220.002, Revision 1
251.005P, Revision 1
251.005, Revision 1
440.091, Revision 1
440.097, Revision 1
440.151, Revision 1
440.152P, Revision 1
440.152, Revision 1
470.002, Revision 1
470.003, Revision 1
470.007, Revision 1
720.058, Revision 1
720.082, Revision 1
720.085, Revision 1
720.095, Revision 1
720.096, Revision 1

DCP/NRC1555

March 17, 2003

Attachment 3

Westinghouse Revised Non-Proprietary Responses
to US Nuclear Regulatory Commission
Requests for Additional Information
dated March 17, 2003

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 220.002 (Response Revision 1)

Question:

For the AP600 containment cylindrical shell, the nominal design thickness is 1.625". However, for the bottom cylinder section, Westinghouse increased the shell thickness to 1.75" in order to "provide margin in the event of corrosion in the embedment transition region" (quote from AP600 DCD). For the AP1000 containment cylindrical shell, the nominal design thickness is a uniform 1.75" for the entire length. The 1.75" thickness just meets the minimum thickness requirements (1.7455") of the 1998 American Society of Mechanical Engineers (ASME) Code Section III, Subsection NE, Paragraph NE-3324.3(a), based on 59 pounds-per-square inch (psi) design pressure, 300°F design temperature, S = 26.4 ksi (thousand pounds-per-square inch), and R = 780". There is no margin in the nominal design thickness for corrosion allowance. Therefore, Westinghouse is requested to provide a technical justification for:

A. eliminating the corrosion allowance for the embedment transition region (deviation from AP600 design philosophy), and

B. making no provision for general corrosion of the containment shell over its 60 year design life in defining the nominal design thickness. Paragraph NE-3121 specifically addresses corrosion allowance for Class MC components.

Westinghouse Response:

A. As stated in the question, Westinghouse increased the shell thickness to 1.75" for the bottom cylinder section in order to "provide margin in the event of corrosion in the embedment transition region". For the AP1000, all of the cylinder section has been increased to 1.75" to accommodate the increase in design pressure. The thickness in the bottom section was not increased further since a greater thickness would require post weld heat treatment.

   The minimum required thickness of the cylinder in accordance with NE-3324.3(a) is 1.726" rather than the 1.7455" stated in the question. This is based on the effect of temperature on allowable stress intensity which is incorrect in the 1995 and 1998 editions of the ASME Code. It has been corrected in the 2002 Addenda. The allowable stress intensity for Class I is 24.3 for all temperatures up to 500 °F. Thus, the allowable stress intensity for Class MC is 26.7 ksi rather than 26.4 ksi. The minimum design thickness for AP600 is 1.597" and the thickness provided is 1.625". Thus the margin for AP1000 is 0.024" compared with 0.028" for AP600.

The containment vessel would be acceptable even if some corrosion were to occur in the transition region. Evaluation would be performed to confirm the structural integrity in accordance with paragraph ASME Section XI, Paragraph IWE-3122.3. The transition region is close to the concrete embedment outside the vessel at elevation 100'. Shell stresses are shown in DCD Figure 3.8.2-5 for the internal design pressure of 59 psig. Table 220.002-1 shows these stresses for locations within 20 feet of the base. The allowable membrane stress intensity away from the discontinuity is 26.7 ksi. Close to the discontinuity the stresses are evaluated as local primary membrane stresses with a 50% higher allowable stress intensity of 40.0 ksi. The maximum stress is 27.30 ksi and occurs at an elevation of 110.0'. This provides significant margin for evaluation of potential corrosion in the transition region.

B. Corrosion protection is provided by coating the vessel as described in DCD subsection 3.8.2-8 and 6.1. Additional margin is provided as described in the response to Part A of this question.

NRC Additional Comments:

A. Westinghouse agreed to provide the following additional information in support of its basis for referencing the 2002 addenda to the ASME code for the AP1000 containment shell design:

(1) technical data in Westinghouse's possession that support the code revision documented in the 2002 addenda, with respect to the acceptability of SA738, Grade B as a containment vessel material;
(2) technical data in Westinghouse's possession (or references to such data) that forms the basis for code revisions from the 1992 edition through the 2002 addenda, with respect to the code-allowable stress intensity values applicable to containment vessel design.

Also, in order to quantify the de-facto corrosion allowance in the embedded transition region of the containment vessel, Westinghouse a complete quantitative assessment of the effect of corrosion in the embedment transition region, considering all applicable load combinations and the associated applicable code stress intensity limits. Local shell bending stress will maximize at the embedment location and will increase as a function of $(tdesign/tcorroded)^2$.

B. Westinghouse agreed to provide the following additional information in support of its basis for relying solely on the containment coatings to prevent general corrosion of the containment vessel:

(1) identify that corrosion prevention is a safety-related function of the coatings on the internal and external surfaces of the containment vessel; and

(2) define the responsibilities of the Combined License applicant for inspection and maintenance of these coatings, in order to preserve the corrosion prevention function of the coatings throughout the unit operating life.

Westinghouse Revised Response:

A. Additional information was provided in revision 1 to the response to RAI 220.003 in support of the basis for referencing the 2002 addenda to the ASME code for the AP1000 containment shell design.

No additional quantification of the de-facto corrosion allowance in the embedded transition region of the containment vessel is required based on the design change described in part B below.

B. Westinghouse is changing the design of the AP1000 containment vessel to add the same corrosion allowance for the embedment transition region as was provided for the AP600. The nominal design thickness for the bottom cylinder section is being increased to 1.875" in order to provide margin in the event of corrosion in the embedment transition region. Since the material thickness will exceed 1.75" , the vertical weld joints in the first course will be post weld heat treated as required by the ASME Code.

Corrosion protection has been identified as a safety related function for the containment vessel coating in DCD subsection 6.1.2.1.1, revision 3. The Combined License applicant will provide a program to monitor the coatings as described in DCD subsection 6.1.3.2.

Design Control Document (DCD) Revision:

Revise fifth paragraph of subsection 3.8.2.1.1 as follows:

The wall thickness **in most** of the cylinder is 1.75 inches. **The wall thickness of the lowest course of the cylindrical shell is increased to 1.875 inches to provide margin in the event of corrosion in the embedment transition region.** The thickness of the heads is 1.625 inches. The heads are ellipsoidal with a major diameter of 130 feet and a height of 37 feet 7.5 inches.

Add a new paragraph in 3.8.2.6 as follows:

The containment vessel is coated with an inorganic zinc coating, except for those portions fully embedded in concrete. The inside of the vessel below the operating floor and up to 8 feet above the operating floor also has a phenolic top coat. Below elevation 100′ the vessel is fully embedded in concrete with the exception of the few penetrations at low elevations (see Figure 3.8.2-4, sheet 3 of 6, for typical details). Embedding the steel vessel in concrete protects the steel from corrosion.

The AP1000 configuration is shown in the general arrangement figures in Section 1.2 and in Figure 3.8.2-1. The exterior of the vessel is embedded at elevation 100′ and concrete is placed against the inside of the vessel up to the maintenance floor at elevation 107′-2″. Above this elevation the inside and outside of the containment vessel are accessible for inspection of the coating. The vessel is coated with an inorganic zinc primer to a level just below the concrete. Seals are provided at the surface of the concrete inside and outside the vessel so that moisture is not trapped next to the steel vessel just below the top of the concrete. The seal on the inside accommodates radial growth of the vessel due to pressurization and heatup.

The plate thickness for the first course (elevation 104′1.5″ to 116′10″) of the cylinder is 1.875 inches, which is 1/8 inch thicker than the rest of the vessel. This provides margin in the event that there would be any corrosion in the transition region despite the coatings and seals described above. Equivalent margin is available for the 1.625-inch-thick bottom head in the transition region (elevation 100′ to 104′1.5″). The plate thickness for the head is a constant thickness and is established by the stresses in the knuckle. As a result, the pressure stresses in the transition zone are well below the allowable stress, providing margin in the event of corrosion in this region.

PRA Revision:

None

Table 220.002-1

Containment Vessel Shell Membrane Stresses

Design Internal Pressure of 59 psig

ELEM	Elev. (ft)	Meridional Stress (ksi)	Circumferential Stress (ksi)	Stress Intensity (ksi)
1	100.00	14.20	4.25	14.20
2	101.02	14.20	2.84	14.20
3	102.05	14.20	1.68	14.20
4	103.09	14.19	3.57	14.19
5	104.13	13.16	8.49	13.16
6	105.10	13.15	14.56	14.56
7	106.08	13.15	19.85	19.85
8	107.06	13.15	23.61	23.61
9	108.04	13.15	25.87	25.87
10	109.02	13.15	26.96	26.96
11	110.00	13.15	27.30	27.30
12	111.00	13.15	27.22	27.22
13	112.00	13.15	26.97	26.97
14	113.00	13.15	26.71	26.71
15	114.04	13.15	26.49	26.49
16	115.08	13.15	26.34	26.34
17	116.12	13.15	26.27	26.27
18	117.16	13.15	26.24	26.24
19	118.20	13.15	26.25	26.25
20	119.24	13.15	26.29	26.29
21	120.27	13.15	26.36	26.36

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.005 (Response Revision 1)

Question:

Provide crack morphology parameters, e.g., surface roughness, number of 45 degree and 90 degree turns, etc., that were used in generating the bounding analysis curves for LBB. To address the staff's concerns resulting from recent experience with stress corrosion cracking in Inconel and stainless steel materials in PWR environments, please provide a comparative study on the most biased line from the LBB candidates using the crack morphology parameters for transgranular stress corrosion cracking. Information regarding crack morphology parameters for various degradation mechanisms is available in NUREG/CR-6443, "Deterministic and Probabilistic Evaluations for Uncertainty in Pipe Fracture Parameters in Leak-Before-Break and In-Service Flaw Evaluations." Report the reduced margin on flaw size from this comparative study of the most biased line when the original bounding analysis curve (BAC) for this line is maintained. (DCD Appendix 3B)

Westinghouse Response:

In generating the bounding analysis curves, crack morphology parameters used are:

> surface roughness = [ ] a,c,e;
> number of 45 degree turns = 0;
> number of 90 degree turns = 0;
> crack shape = rectangle.

In order to avoid the concern of the Inconel 82/182 PWSCC issue for the AP1000, Inconel 52/152 will be used for all applicable locations of LBB piping systems.

Westinghouse believes that the Transgranular Stress Corrosion Cracking (TGSCC) mechanism is highly unlikely in the AP1000 LBB candidate piping systems. TGSCC has not been observed in PWR stainless steel piping. For TGSCC to occur, an aggressive species such as chlorides would also need to be present. Since these types of aggressive species will be controlled and kept at minimum levels in the AP1000 LBB candidate piping systems water environment, a much higher level of oxygen would be required to be present to provide the appropriate environment for SCC to develop. Since the oxygen levels are kept to near zero by the hydrogen overpressure, the AP1000 piping systems will not be susceptible to TGSCC.

The occurrence of TGSCC in CRDMs at Palisades and Ft. Calhoun (basically the same CRDM design) is not expected in AP1000 LBB candidate piping systems. The Palisades incidents occurred because the materials used were susceptible to TGSCC in the CRDM environment (elevated levels of dissolved oxygen, some level of chloride ions. The TGSCC cracking incidents at Palisades and Ft. Calhoun CRDM are unique to that geometry and do not apply to AP1000.

Since the AP1000 piping systems will not be susceptible to TGSCC, we do not believe a leak rate calculation based on the hypothetical assumption of TGSCC for the AP1000 LBB application is necessary.

NRC Additional Question

As discussed in a teleconference with Westinghouse and NRC on March 10, 2003, in order to perform confirmatory calculations related to RAI 251.005, the staff requests the additional information. Please provide the following information related to Figure 3B-12:

(1) Young's modulus,
(2) 0.2% offset yield stress,
(3) Ultimate tensile strength,
(4) Ramberg-Osgood exponent alfa and n specified for elastic response,
(5) the angle beta that defines the neutral axis of the cracked section at Point 1,
(6) the calculated elastic and plastic crack opening displacement s,
(7) entrance loss coefficient,
(8) the ratio of length for single-phase region L to hydraulic diameter D,
(9) friction factor for the momentum equation, and
(10) the ratio of the crack exit to inlet area.

Westinghouse Additional Response:

The information requested is provided in the attached table.

Design Control Document (DCD) Revision:

None

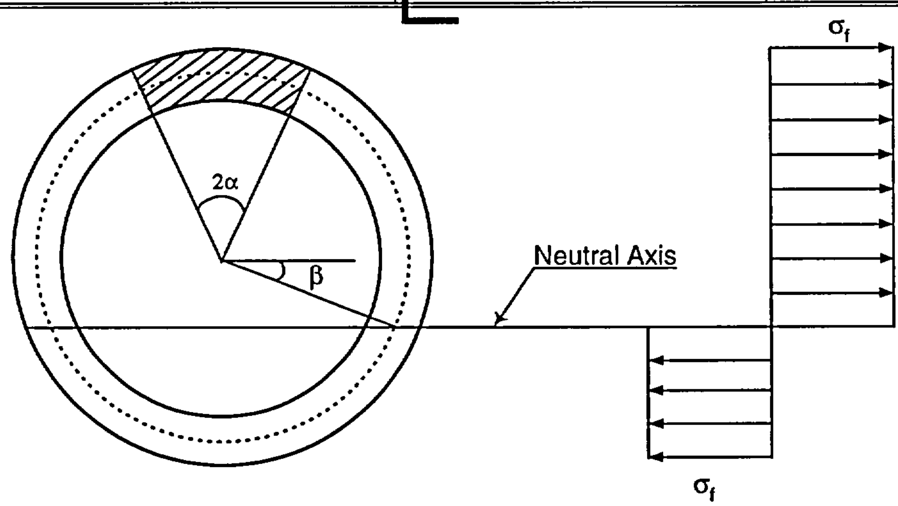
PRA Revision:

None

Table 251.005-1

ITEM	LOW NORMAL CASE	HIGH NORMAL CASE
Outer Diameter	12.750 in.	12.750 in.
Thickness	1.169 in.	1.169 in.
Pressure	2262.7 psia	2262.7 psia
Temperature	610.0 ^{0}F	610.0 ^{0}F
Young's Modulus	25.25x10^6 psi	25.25x10^6 psi
0.2% offset Yield Strength	18200 psi	18200 psi
Ultimate Tensile Strength	63200 psi	63200 psi
Flow Stress	40700 psi	40700 psi
Ramberg-Osgood Exponent Alpha & N specified for elastic response		
The angle Beta that defines the neutral axis of the cracked section at point 1		
The calculated elastic & plastic crack opening displacements		
Entrance loss coefficient		
The ratio of length for single-phase region L to hydraulic diameter D		
Friction Factor for the momentum equation		
The ratio of the crack exit to inlet area		

a,c,e



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.091 (Response Revision 1)

Question:

In the discussion and analysis of the double-ended direct vessel injection (DEDVI) line break (Section 15.6.5.4C.2), it was assumed that the ADS4A valve failed to open (single failure) and the containment pressure is at the WGOTHIC calculated minimum. These conditions may be conservative for depressurization but not from the point of view of long-term cooling. Consider the case when all ADS4 valves open, with a maximum containment pressure. Steam velocity in the ADS4s will be minimum.

Will that steam velocity be able to entrain and remove liquid from the core? (Note, it is not feasible to draw this conclusion from the information in the code applicability report without extensive calculations.)

Westinghouse Response:

The question is concerned with the removal of liquid from the AP1000 core / upper plenum at a minimum steam velocity condition. The concern is whether there exists sufficient liquid carryover out the ADS-4 valves post-LOCA to prevent the potential for boron precipitation in the reactor vessel. As was demonstrated for the AP600, liquid carryover from the ADS-4 valves prevents the reactor vessel boron concentration from approaching the concentration where boron would precipitate. A boron concentration of approximately 35,000 ppm at 240F is necessary to cause boron to precipitate out of solution. Bounding calculations performed for the AP1000 have determined that the maximum post-LOCA boron concentration calculated for the limiting long-term cooling analysis cases presented in the DCD is 5500 ppm. This represents a large margin to the value where boron precipitation could be a concern.

This RAI asks that a less conservative (from a core cooling perspective) long-term cooling analysis be performed, to assess whether reduced ADS-4 liquid carryover will result, and thus a higher potential for post-LOCA boron precipitation. A WCOBRA/TRAC calculation of the DEDVI line break has been performed to investigate this scenario. In order to minimize the ADS-4 steam velocity, all the ADS-4 valves are assumed to open as requested in the RAI. In addition, containment pressure is set equal to the maximum calculated pressure from the containment integrity analysis reported in Chapter 6 of the AP1000 DCD. This containment pressure is calculated for a double-ended RCS loop pipe rupture using assumptions that maximize the calculated pressure result; it identifies an upper bound to the pressure response anticipated for a DEDVI break. The DEDVI break features the early actuation of ADS-4, which results in flow through ADS stages 1-3 being limited to a short time interval, thus minimizing IRWST water heatup prior to the long-term cooling phase. Therefore, there will be maximum subcooling of the injection water entering the downcomer during long-term cooling, so a minimum amount of steam is generated in the core.

A set of figures is provided presenting the results obtained from the WCOBRA/TRAC run for this scenario. Liquid flow through the ADS-4 flow paths is adequate to ensure that boron will not concentrate in the core. As seen on Figure 440.091-15, the average steam velocity through the ADS-4 Stage 4A flow path in the offtake pipe from the hot legs is 95 ft/sec, while in the DCD case the average steam velocity at the same location is almost 300 ft/sec (Figure 440.091-16). Overall, the figures show that amount of liquid carryover is increased for this scenario. As seen in comparison to the DEDVI long-term cooling analysis results presented in the DCD, the vessel injection is greater, and reactor vessel level is higher throughout the transient. This is primarily due to the faster RCS depressurization resulting from all four ADS-4 valves opening. Even though the vapor velocity in the ADS-4 flow paths is reduced, the core remains covered and cooled, and the liquid carry-over out the ADS-4 is increased (when compared to the cases presented in the DCD). Thus boron precipitation in the reactor vessel will not be a concern for the postulated scenario raised in this RAI.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

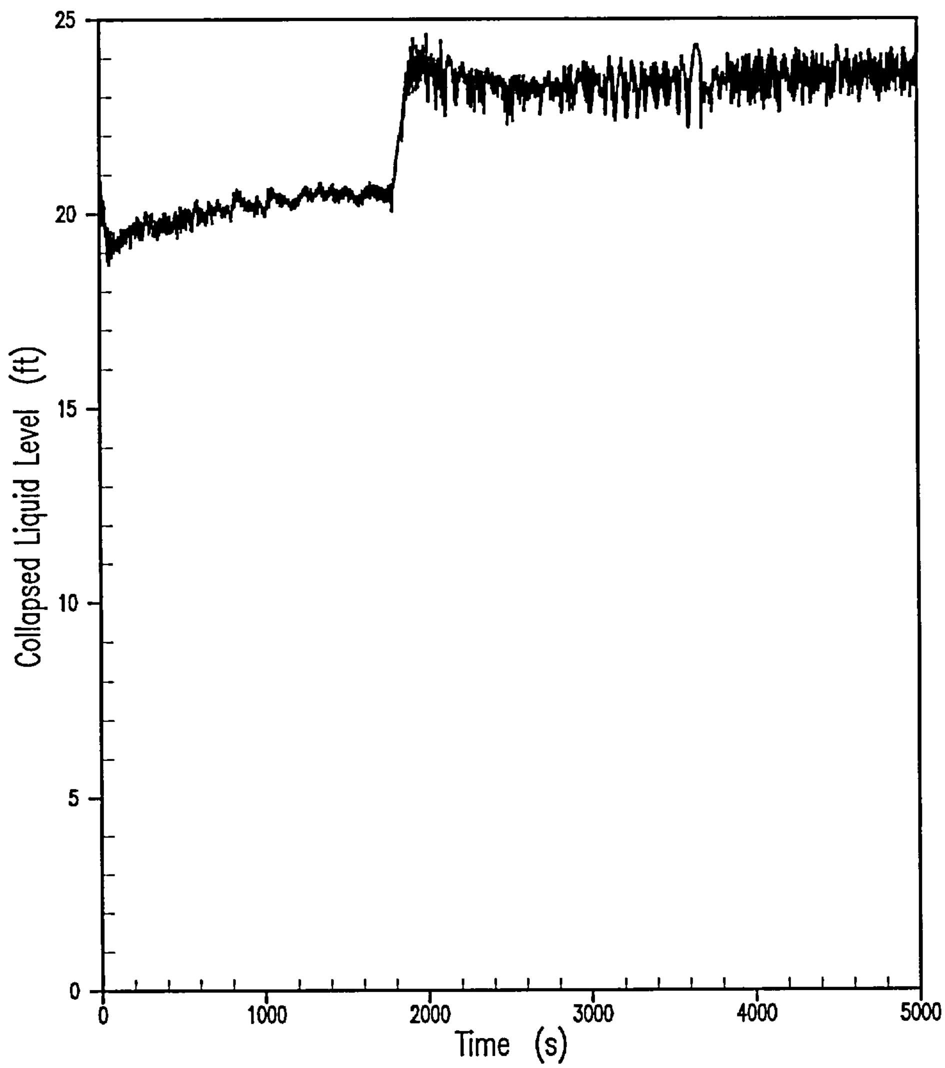


Figure 440.091-1 **Collapsed Liquid Level in the Downcomer**

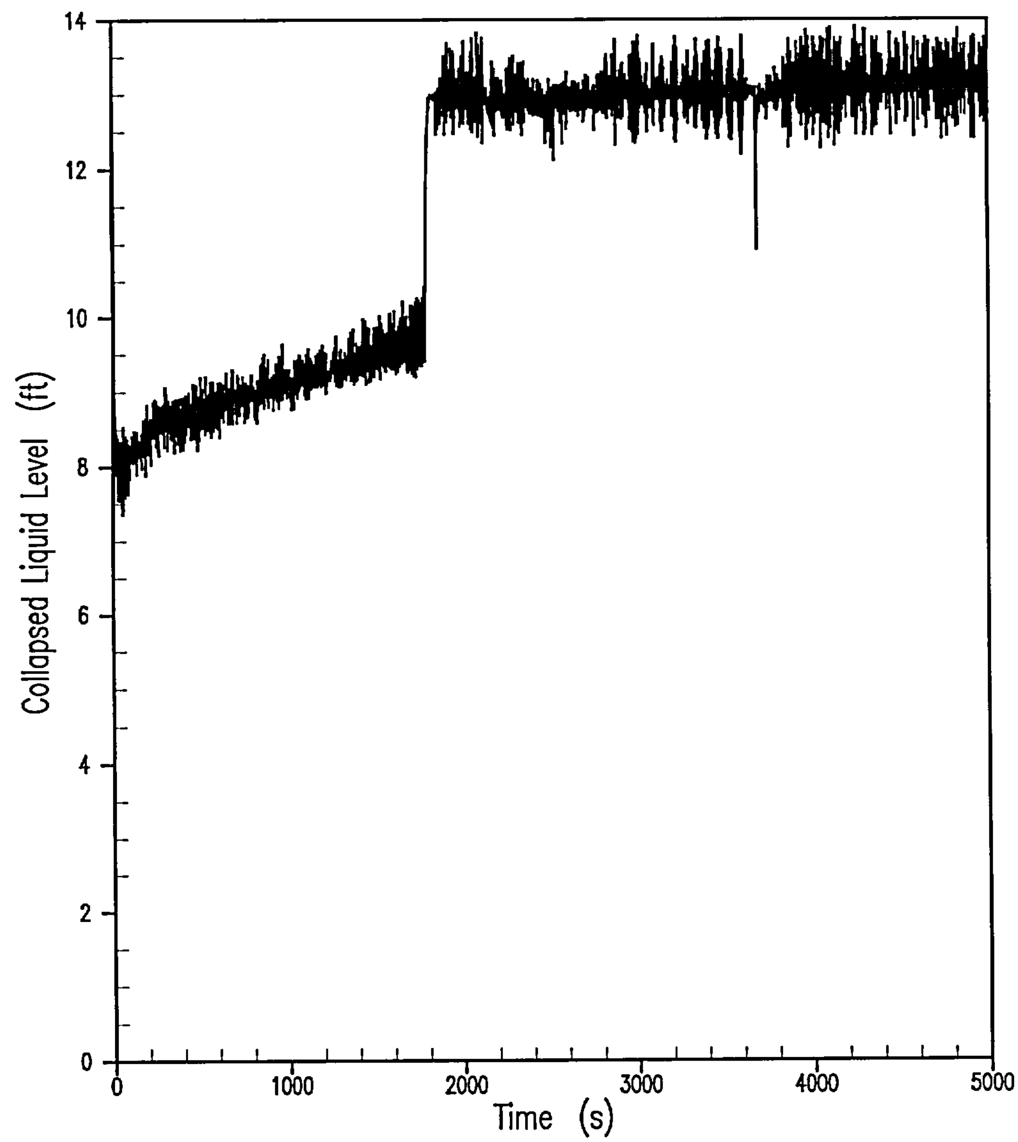


Figure 440.091-2 Collapsed Liquid Level Over the Heated Length of the Fuel

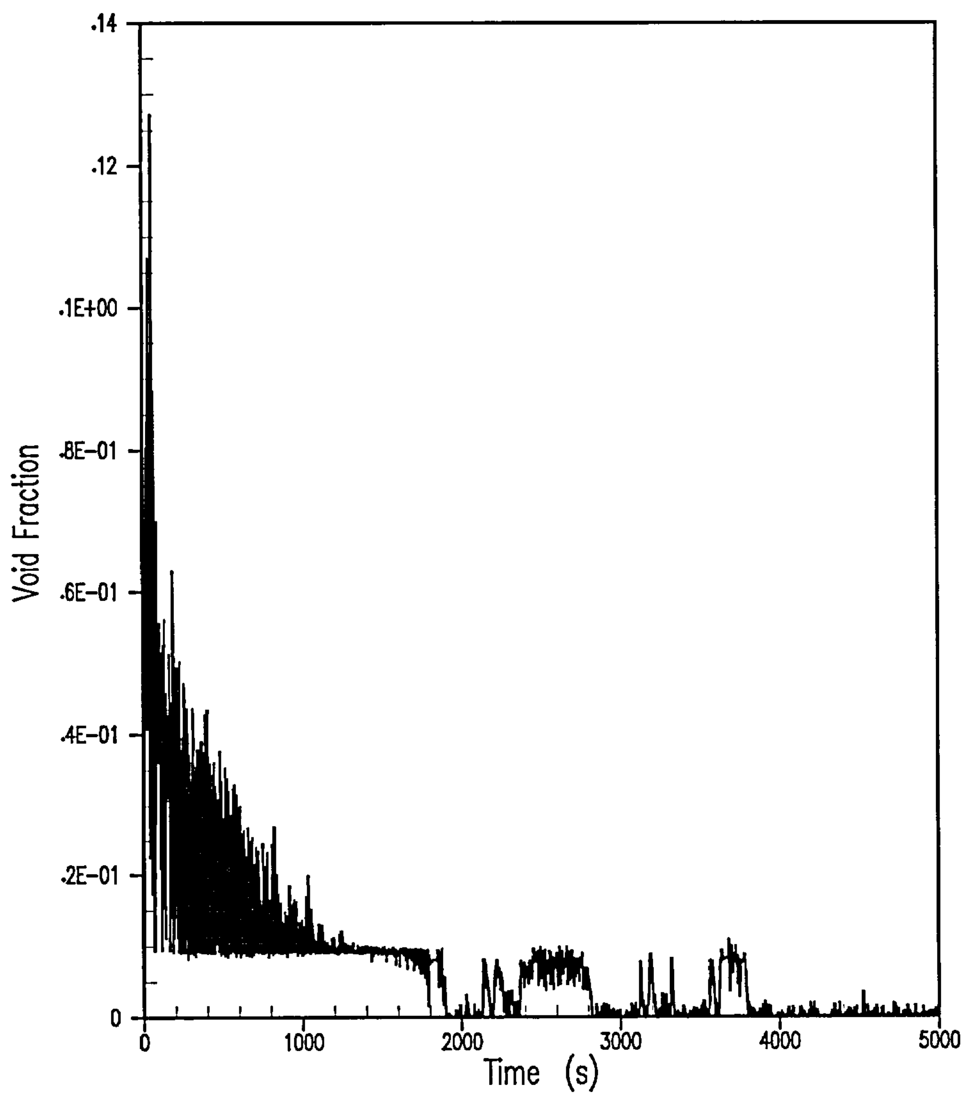


Figure 440.091-3 Void Fraction in Core Cell Level 1 of 2

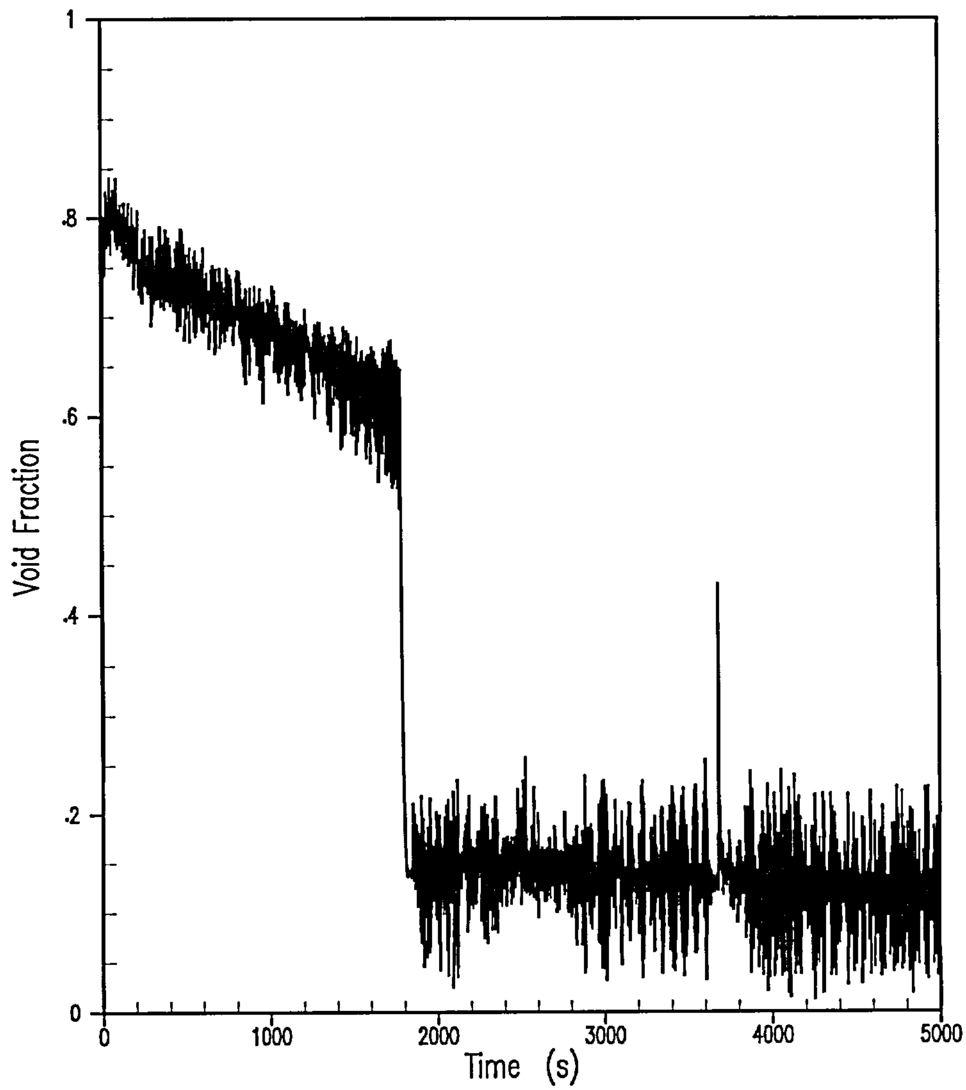


Figure 440.091-4 Void Fraction in Core Cell Level 2 of 2

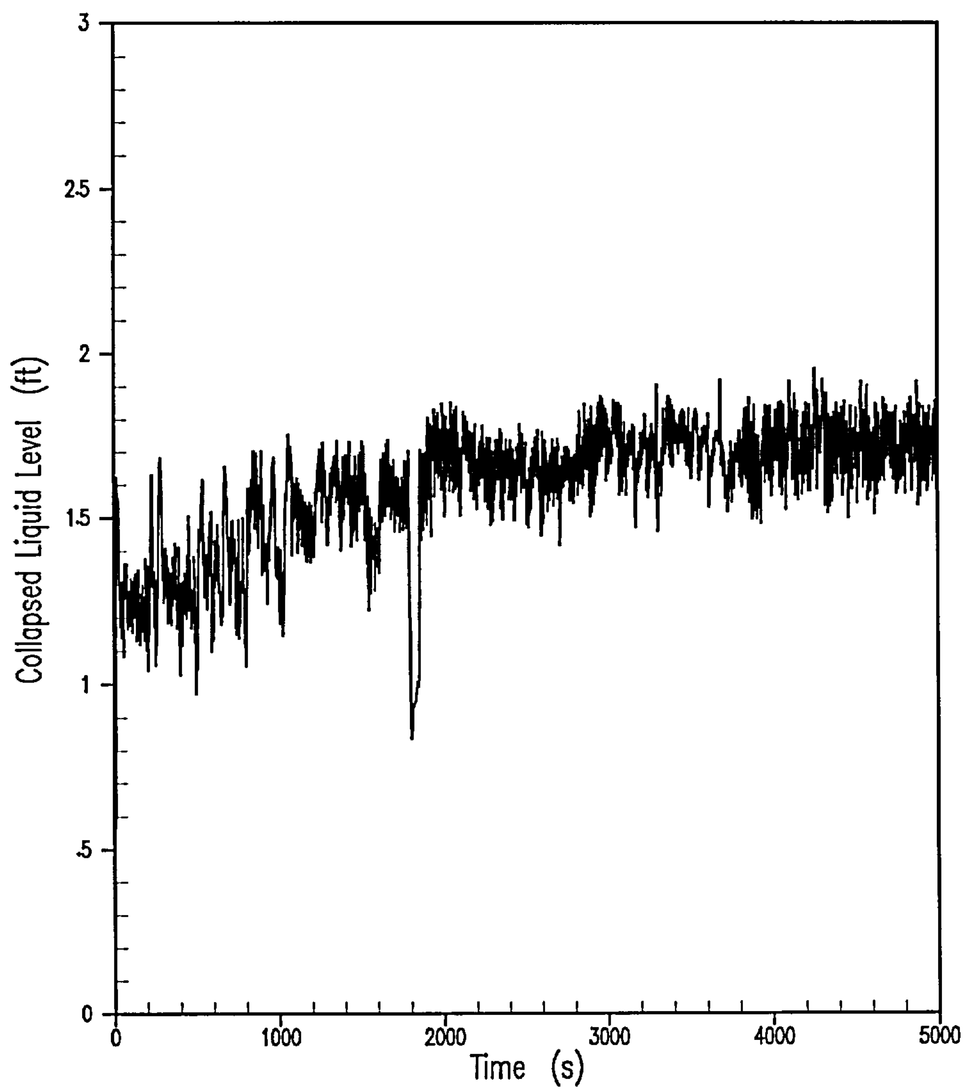


Figure 440.091-5 Collapsed Liquid Level in the Hot Leg of Intact Loop

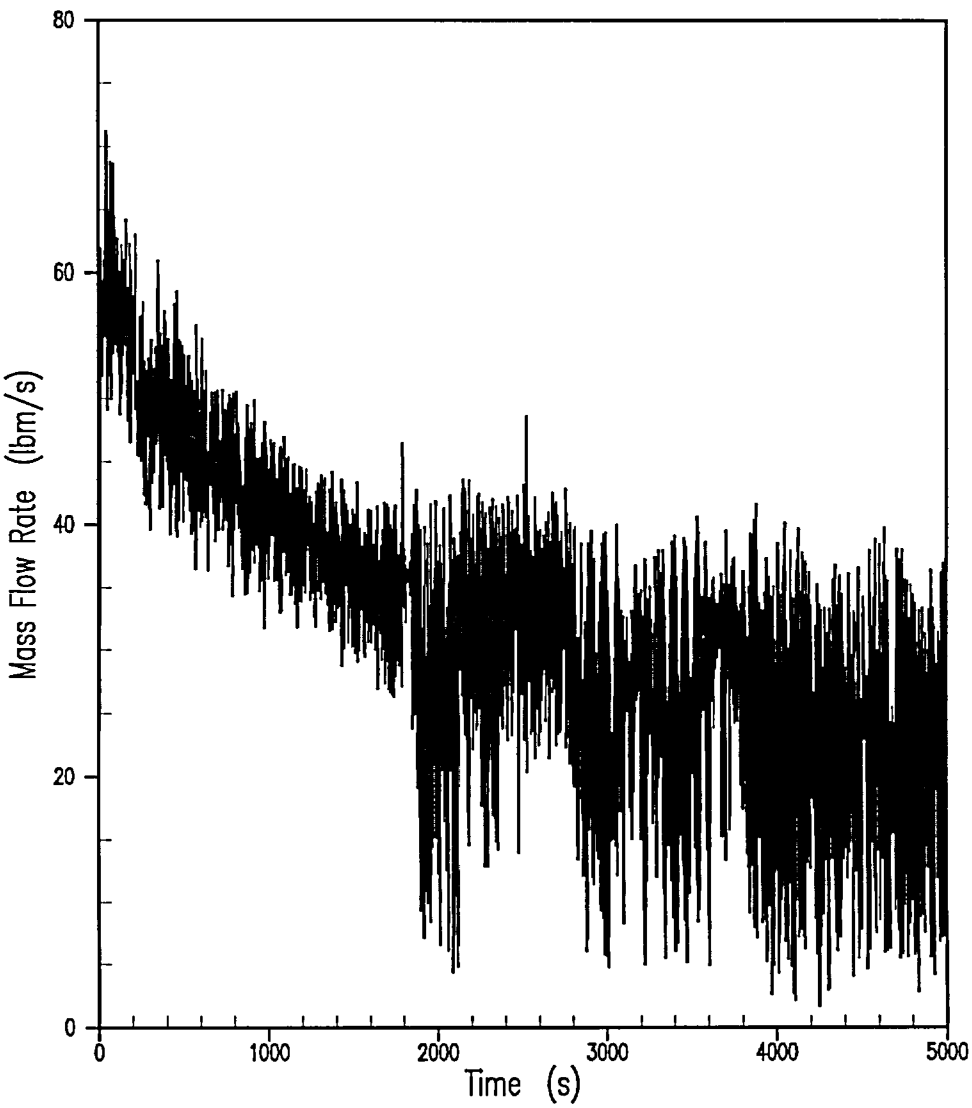


Figure 440.091-6 Vapor Rate out of the Core

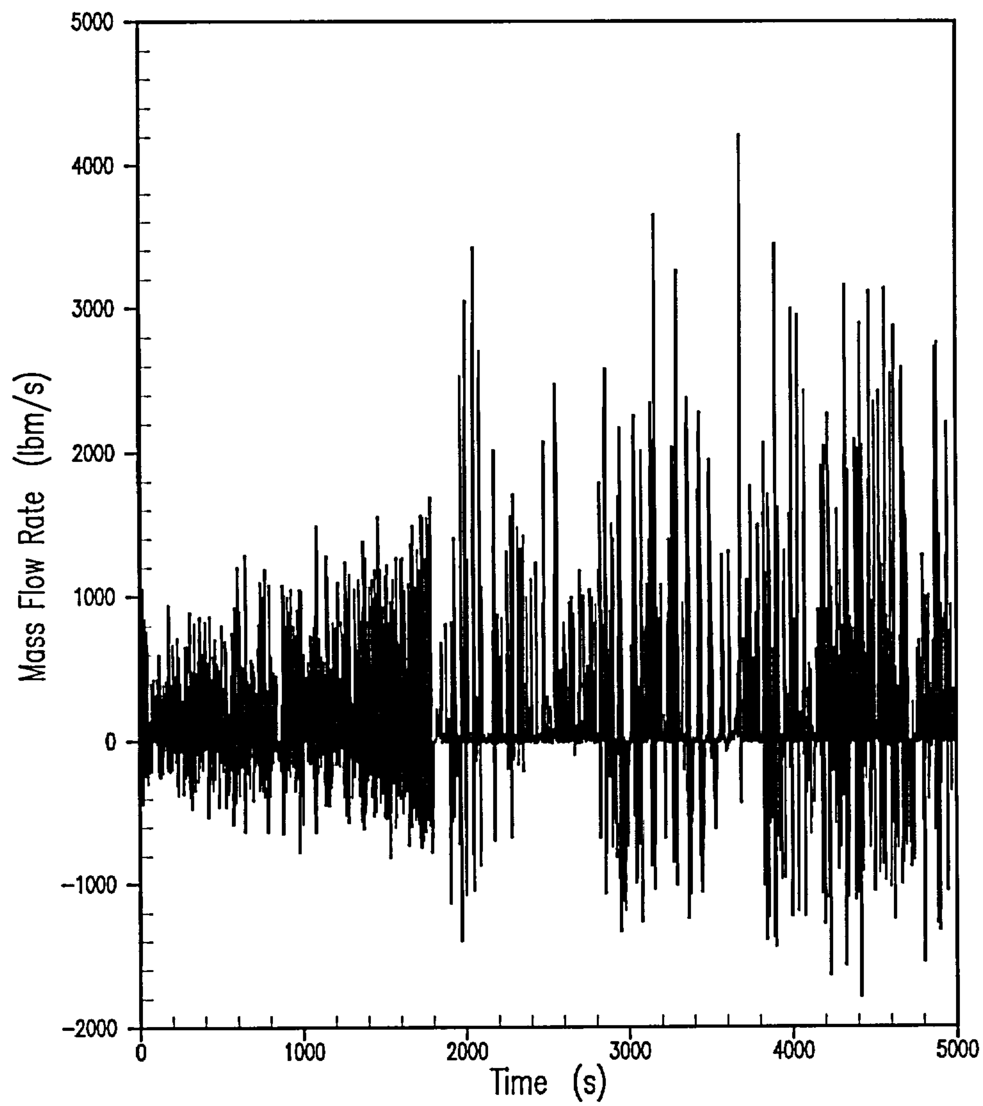


Figure 440.091-7 Liquid Flow Rate out of the Core

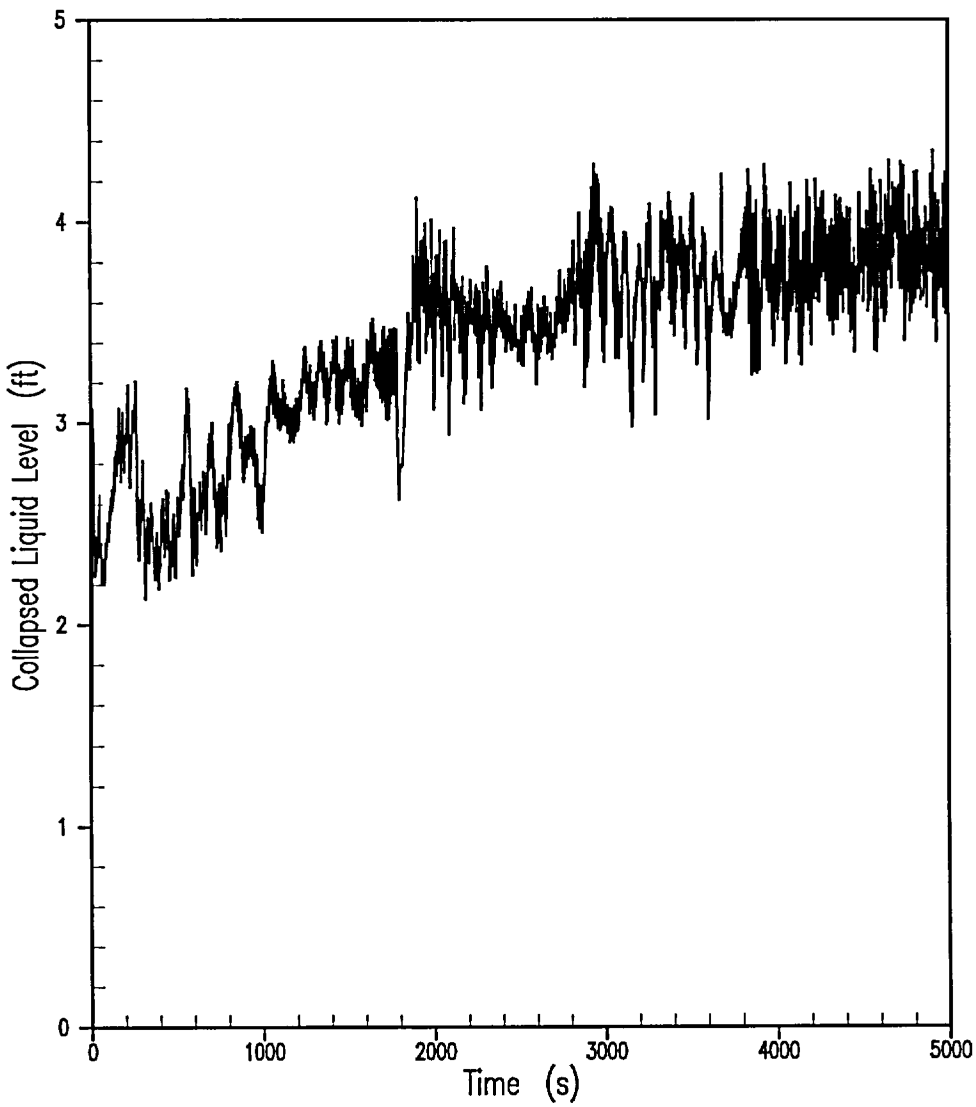


Figure 440.091-8 Collapsed Liquid Level in the Upper Plenum

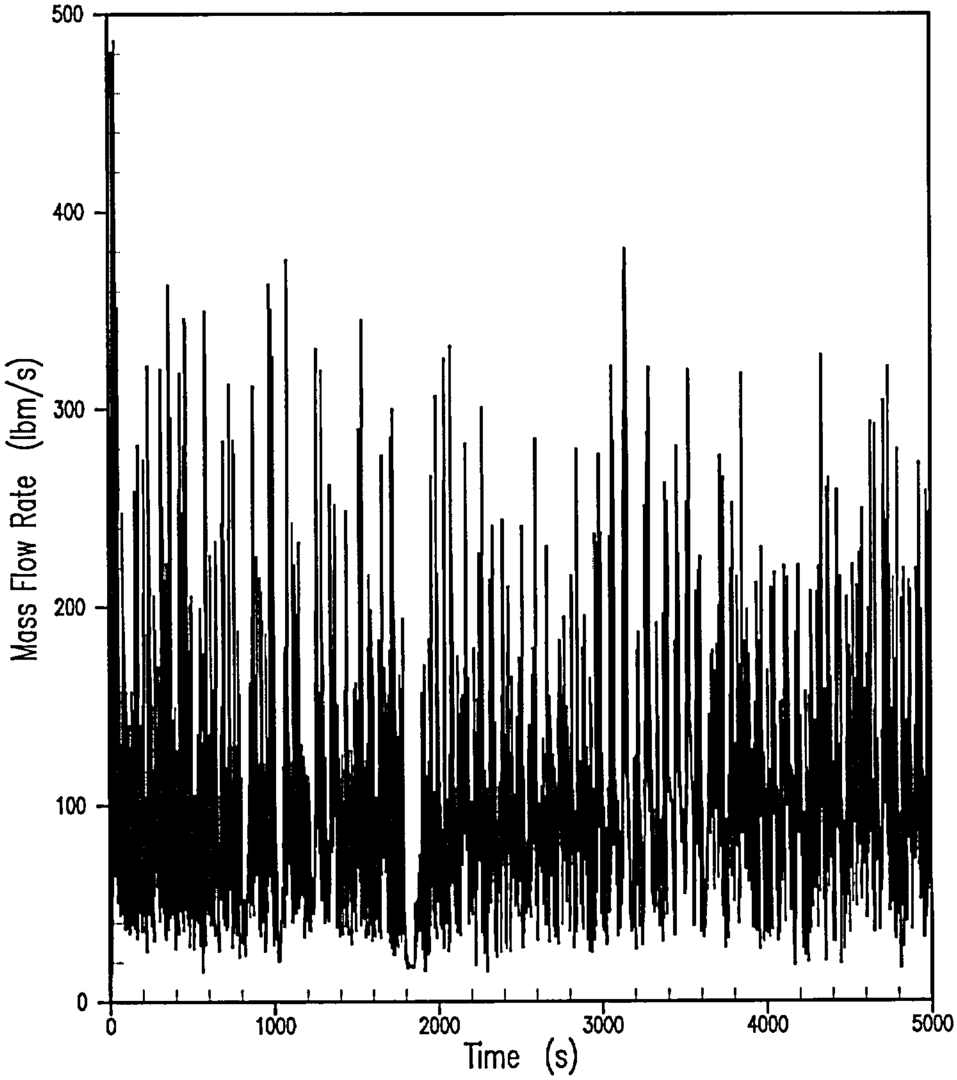


Figure 440.091-9 Mixture Flow Rate Through ADS Stage 4A Valves

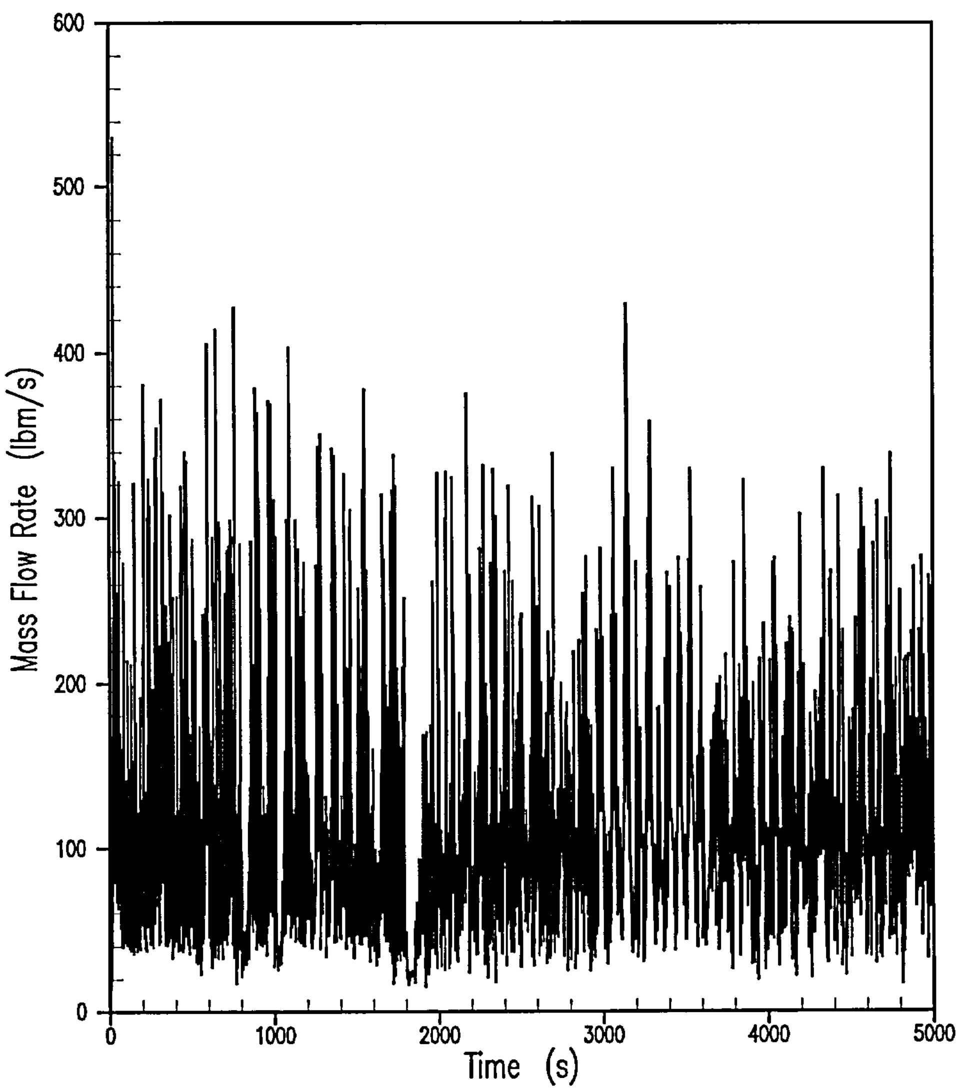


Figure 440.091-10 Mixture Flow Rate Through ADS Stage 4B Valves

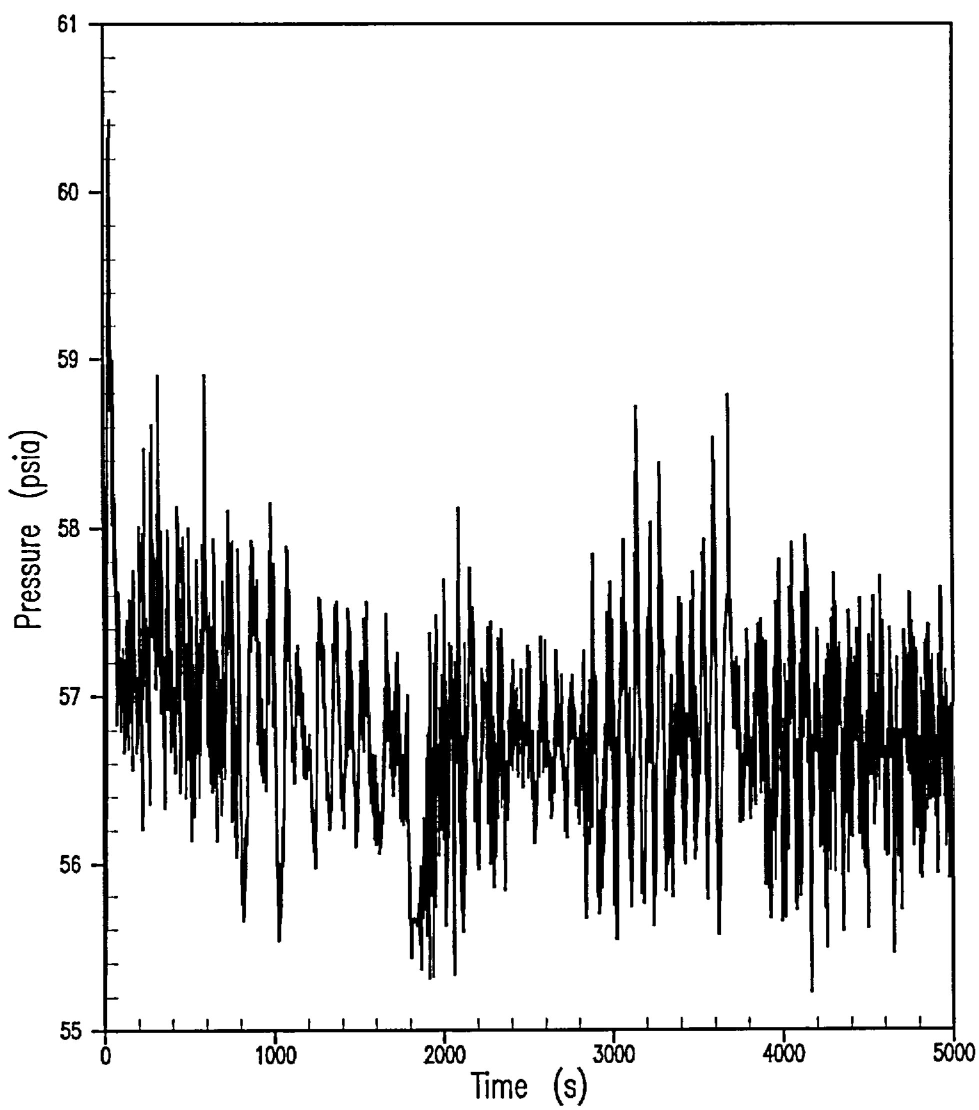


Figure 440.091-11 Upper Plenum Pressure

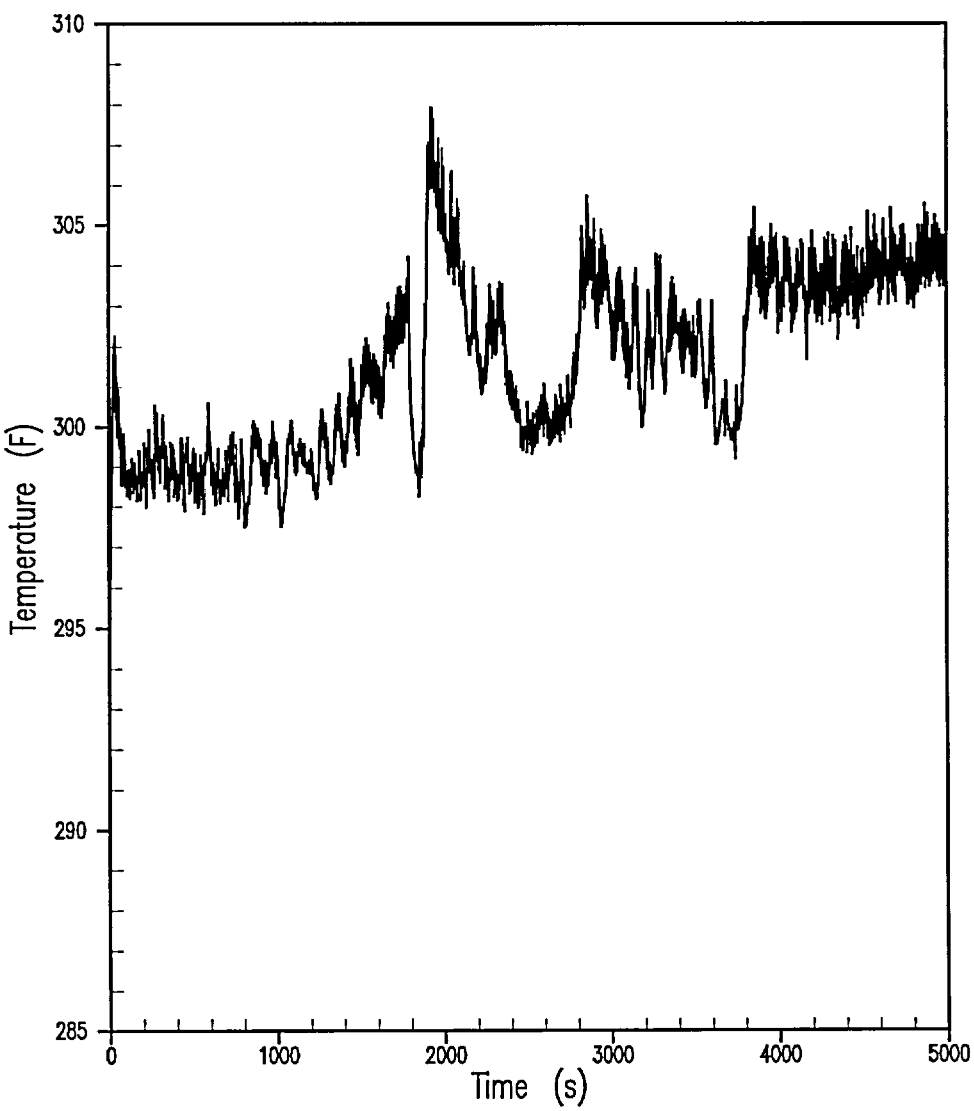


Figure 440.091-12 PCT of the Hot Rod

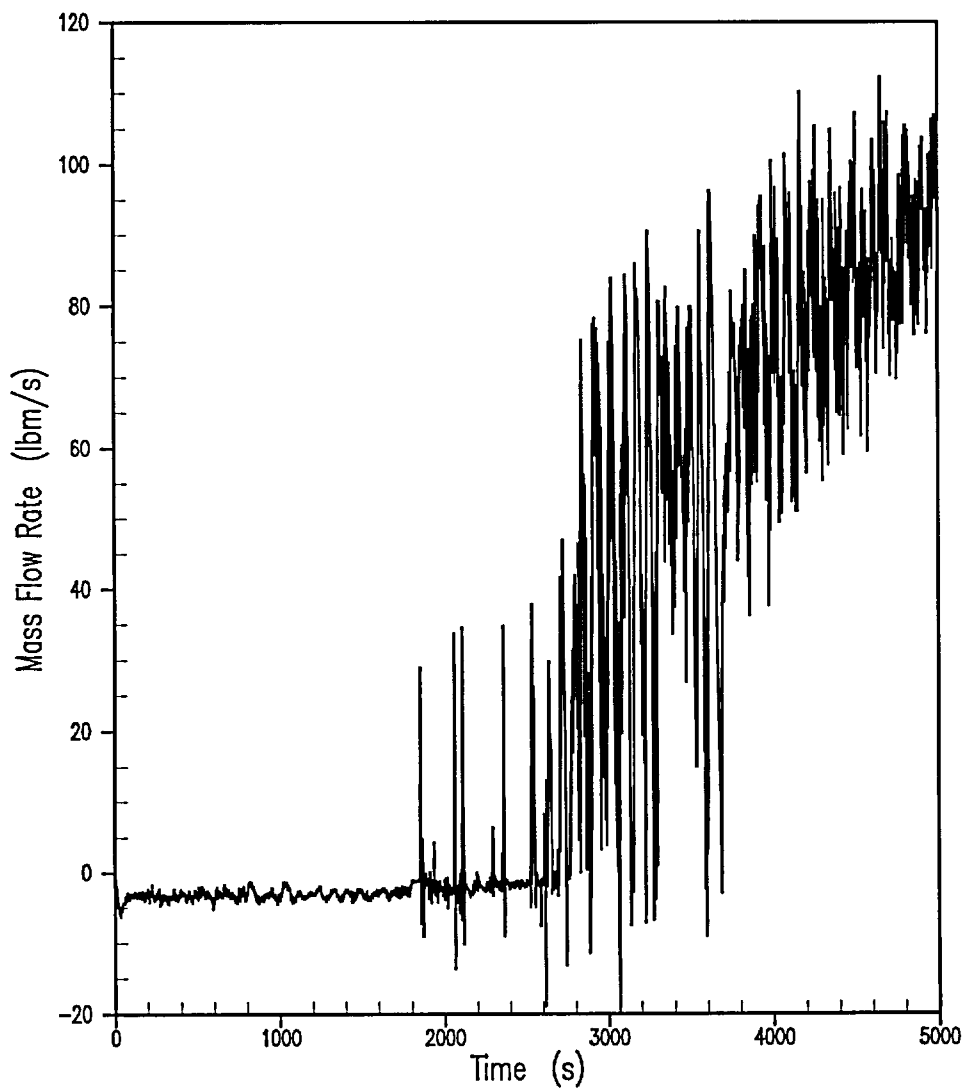


Figure 440.091-13 DVI-A Mixture Flow Rate

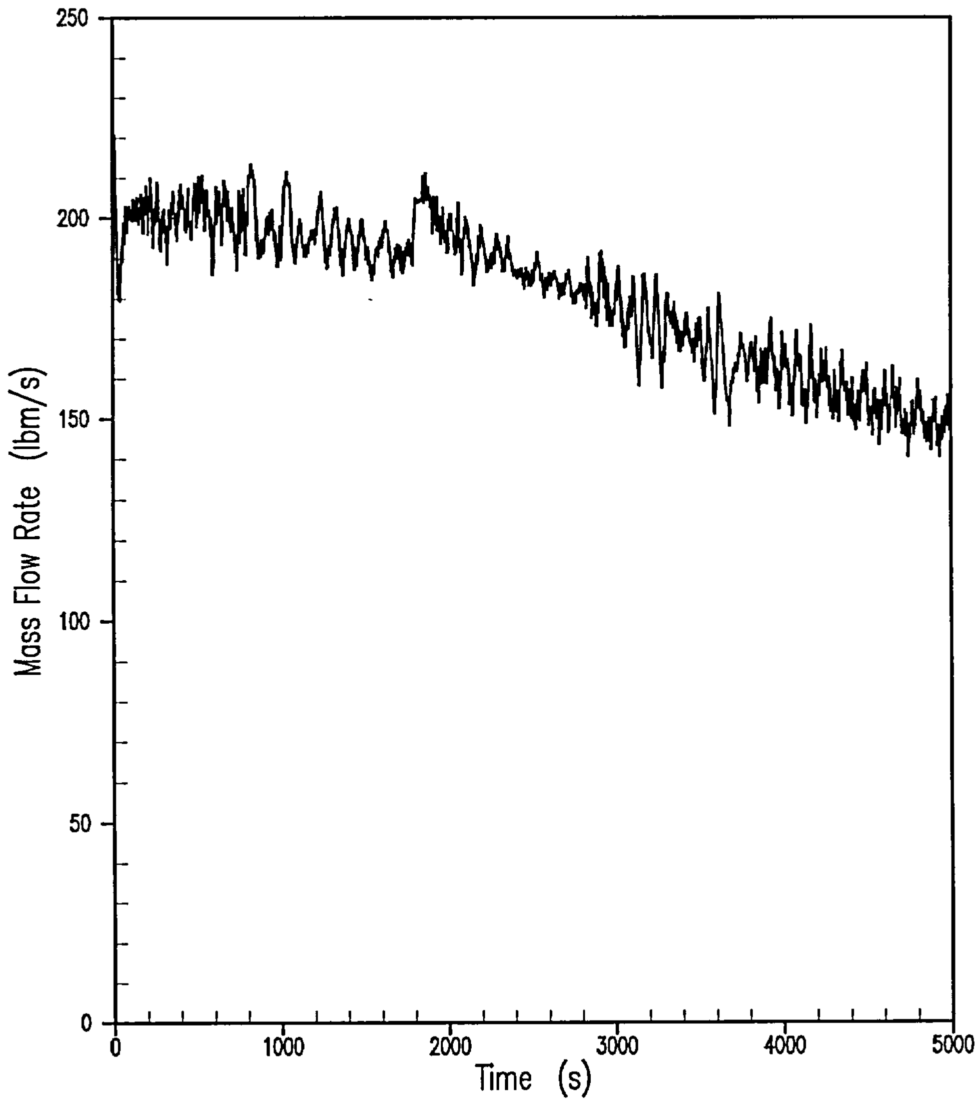


Figure 440-091-14 DVI-B Mixture Flow Rate

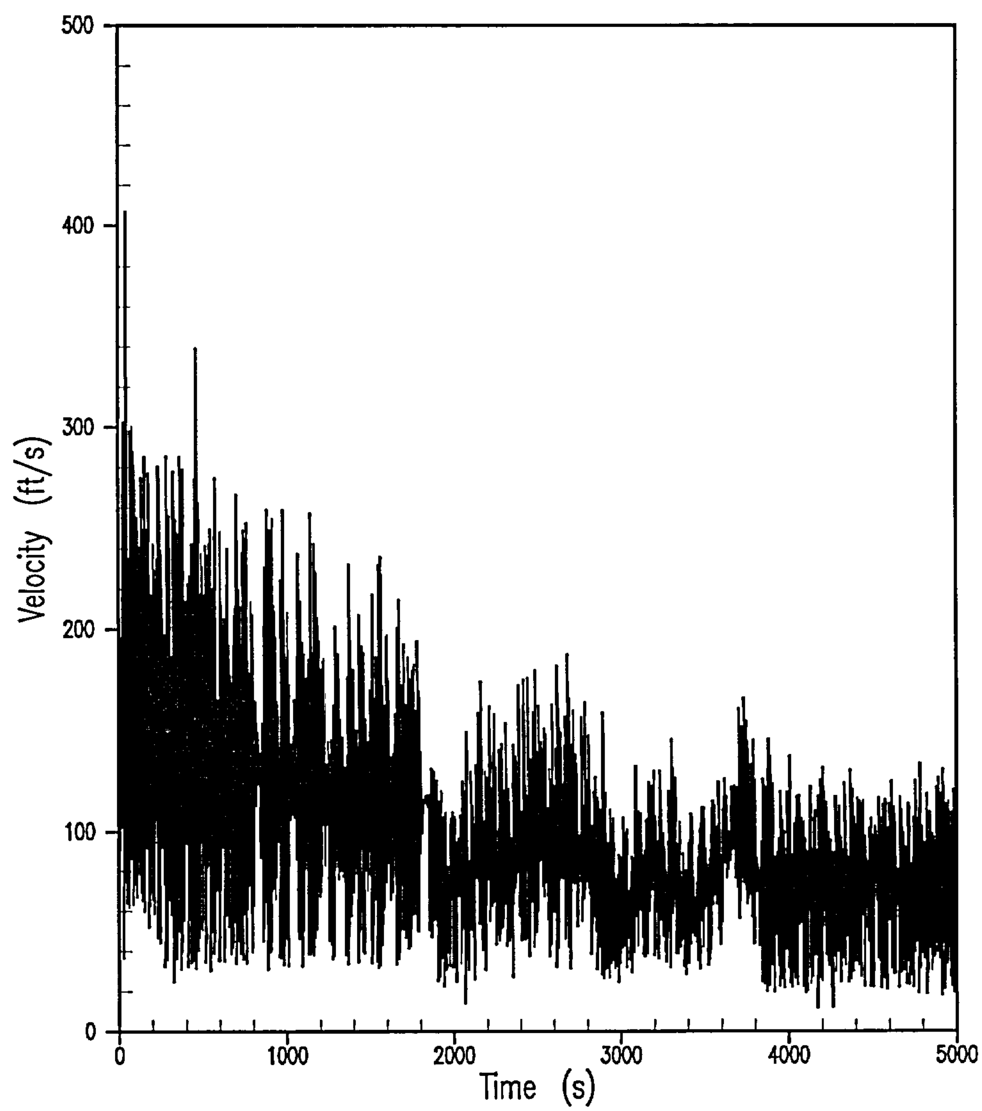


Figure 440.091-15 ADS Stage 4A Vapor Velocity at Offtake Pipe

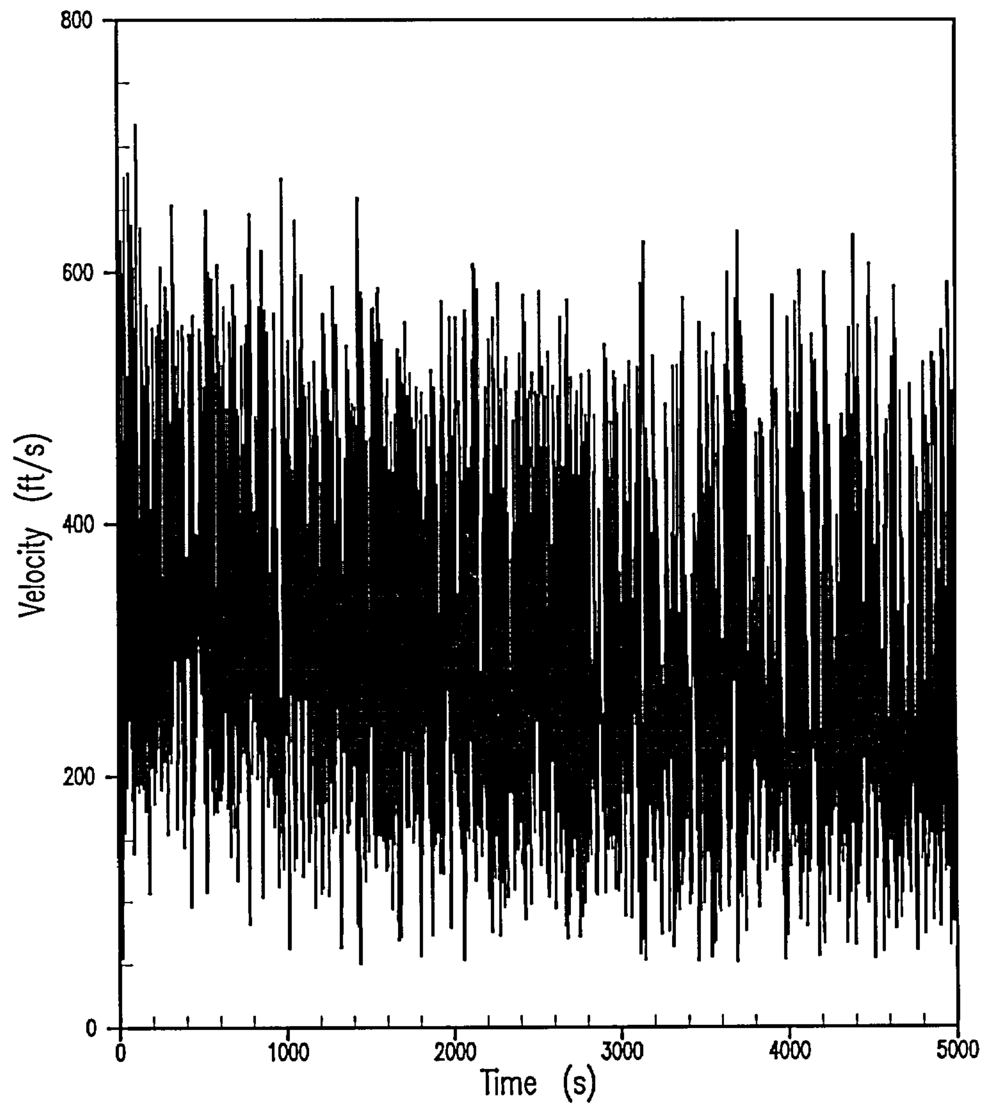


Figure 440.091-16 ADS Stage 4A Vapor Velocity at Offtake Pipe (DCD Long-term Cooling Case)

NRC Additional Comments:

The response did not address the question i.e. given the differences in the physical dimensions between AP600 and AP1000 is WCOBRA/TRAC qualified to calculate liquid entrainment in the ADS-4 under conservative long-term cooling (LTC) conditions. The response does not allow a quantification of the liquid flow.

Please address the following: (1) the qualification of WCOBRA/TRAC to calculate liquid entrainment in the automatic depressurization system stage 4 (ADS-4) under conservative LTC conditions for the physical configuration of the AP1000, (i.e. slug flow in AP600 vs droplet entrainment in AP1000, (2) is 95 ft/sec above the entrainment velocity for the AP1000 configuration? And (3) quantify the liquid flow through ADS-4 for the conditions discussed in the December 2, 2002 response.

Westinghouse needs to address flow regimes and include ADS-4 liquid mass flow rate

Westinghouse Revised Response:

WCOBRA/TRAC has been validated for the calculation of advanced plant long-term cooling phenomena against APEX facility data in WCAP-14776, Rev. 4. The attached figures 440.091-1-1 and 440.091-1-2 show that the validation was performed for ADS-4 two-phase flow data in the annular flow regime, and that flow regime is present in the ADS-4 piping for the scenario requested in this RAI. The uncircled "+" symbol on each figure shows the flow regime from the run performed for this RAI response. Annular flow is predicted to occur in both the vertical and horizontal piping segments of the ADS-4 piping during the RAI-440.091 scenario, as it is in the AP1000 DCD Section 15.6.5.4C DEDVI break analysis. Since annular flow is the flow regime present in the APEX test ADS-4 piping and also in both of these AP1000 simulations, in each case the velocity at the start of sump injection is adequate to move liquid through the vertical and horizontal ADS-4 piping lengths until discharge into containment. The AP1000 long-term cooling ADS-4 flow path flow regime is concluded to correspond phenomenologically to the APEX test condition with or without the postulated single active failure of an ADS-4 valve.

The onset of droplet entrainment in annular vertical gas/ liquid flow for the scenario in this RAI has been identified using the dimensionless gas velocity criterion of Reference 440.191-1. The velocity at which droplet entrainment occurs is calculated to be approximately 50 ft/sec using this correlation. Therefore, liquid entrainment will occur in the AP1000 ADS-4 piping during long-term cooling.

The liquid discharge rate through the four open ADS-4 flow paths is about 180 lbm/sec for the scenario of this RAI. This is approximately 28 lbm/sec greater than the ADS-4 liquid flow discharge through the three open flow paths in the reference DCD Section 15.6.5.4C long-term cooling case.

Reference

Wallis, G. B., "Phenomena of Liquid Transfer in Two-Phase Dispersed Annular Flow," Int. J. Heat Mass Transfer, Vol. 11, pp. 783-785, 1968.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

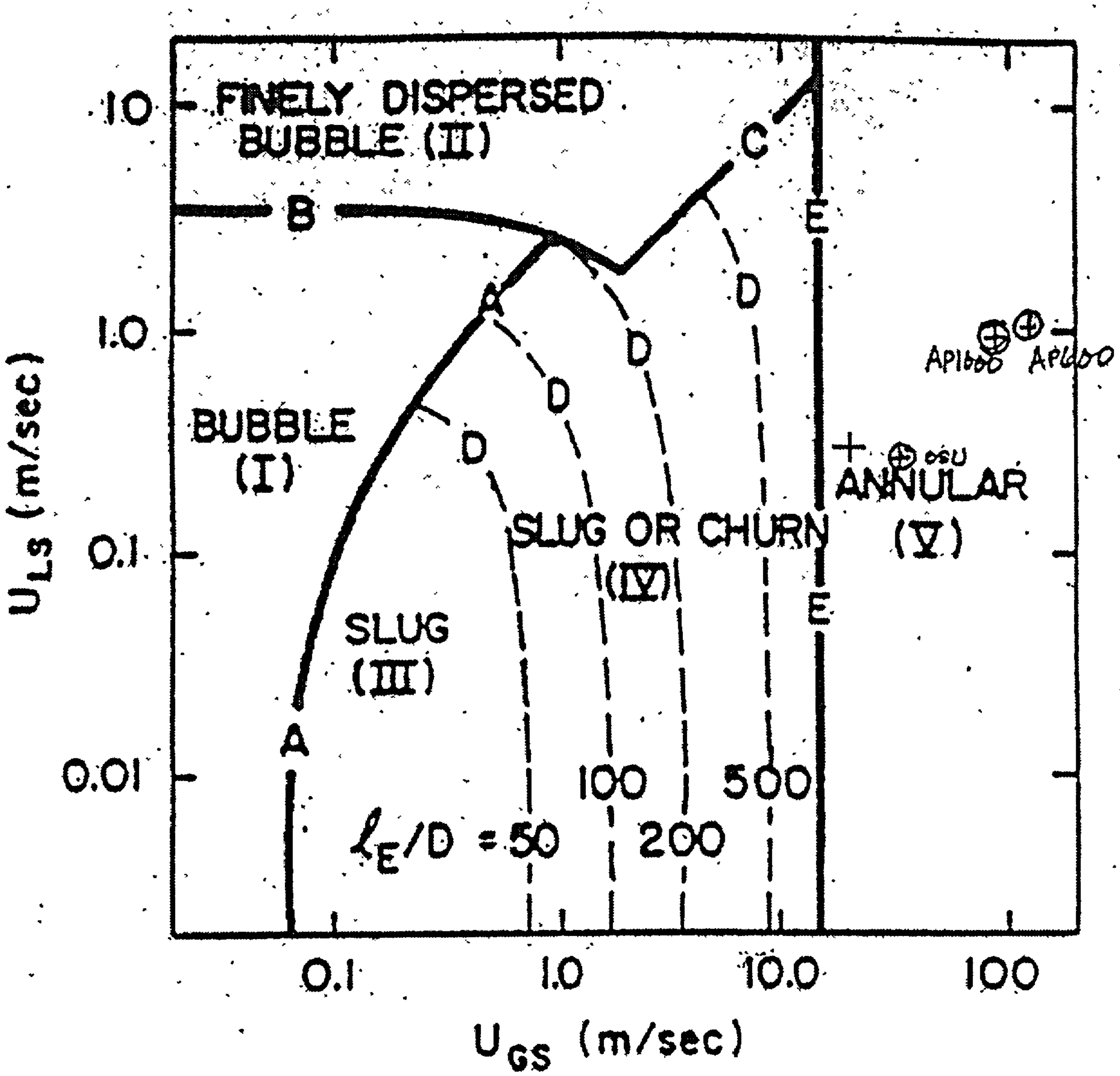


Figure 440.091-1-1 Sump Injection Phase – ADS4 Pipe (Vertical) Vertical Flow Regime Map (Taitlel-Dukler)

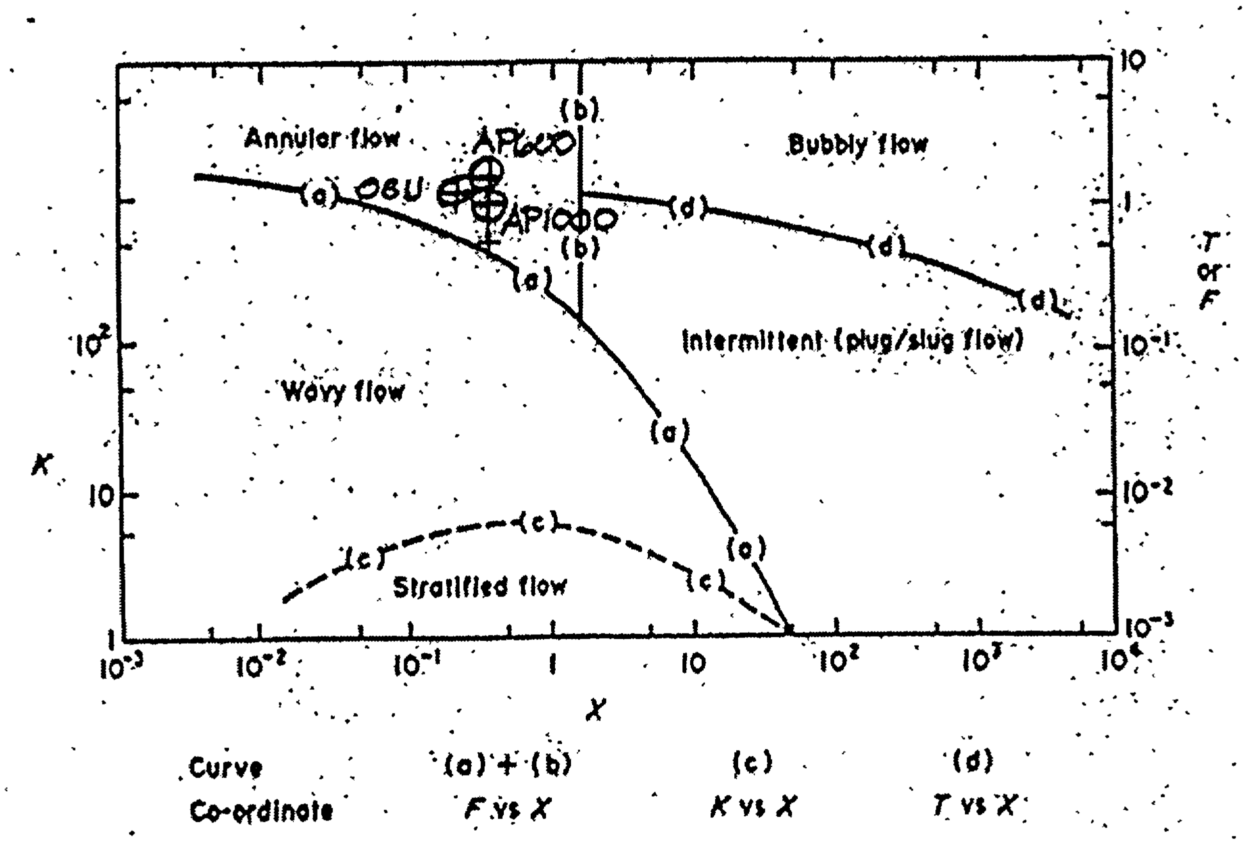


Figure 440.091-1-2 Sump Injection Phase – ADS4 Pipe (Horizontal) Horizontal Flow Regime Map (Taitlel-Dukler)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.097 (Response Revision 1)

Question:

The documentation of the large break LOCA (LBLOCA) analysis methodology and results in Section 15.6.5.4A is totally inadequate.

Please include additional information comparable in content and detail to the small break LOCA (SBLOCA) and the long-term cooling.

NRC Additional Comments:

In your response you state that "Reference 3 indicated the application restrictions on the AP600 methodology. The AP1000 large break LOCA analysis has complied with those restrictions." Because the estimated peak cladding temperature (PCT) is higher than 1725□F you must address limitation 4. You indicate that you addressed the global model matrix (Limitation 4a) and the sensitivity to the modeling of the CMT and PRHR (limitation 4b).

A. Please address limitations 4c and 4d, i.e., maximum local oxidation and submit the results for staff review.

B. The integrals of the flows shown in Figures 15.6.5A-5 and -6 do not match the contents of the accumulators and the core makeup tanks (CMTs), respectively, shown in Table 2.1-1 of WCAP-15612. Please comment on the flows shown in these figures and indicate in a single graph the core coolant inflow, outflow, downcomer level and core level vs time from the initiation of the transient to the initiation of in-containment refueling water storage tank (IRWST) injection (a similar graph was prepared for AP600).

C. What is the assumed single failure in the LBLOCA analysis and what is the exact calculated value of the maximum cladding oxidation? (Table 15.6.5-8)

D. It is stated (pg. 15.6-46a) that "Figure 15.6.5A-12 presents the collapsed liquid levels in the core referenced to the bottom elevation of the active fuel (solid line) and downcomer (dashed line) referenced to the bottom of the reactor vessel". How is it possible that these levels have the same value at about 20 seconds?

E. Best estimate LBLOCA analyses assume an unfavorable flow location for the hot assembly (even if the actual hot assembly is not in such location in the configuration being analyzed). Was this implemented in this analysis?

Westinghouse Response: (Revision 1)

The large break LOCA results described in Section 15.6.5.4A were significantly expanded for DCD Chapter 15 Revision 3. The following addresses NRC staff comment items A-E above.

A. The large break LOCA peak cladding temperature is calculated to be 2124F at the 95th percentile. To demonstrate compliance with the 10CFR50.46 Acceptance Criteria, the local cladding oxidation is computed according to the methodology approved for use in 3/ 4 loop plant applications as found in Volume 5, Section 26-5-3-1 of WCAP-12945-P-A. The core-wide oxidation is computed according to the approved methodology described in Volume 5, Section 26-5-3-2 of WCAP-12945-P-A.

The two oxidation calculations also address Limitation 4c of the application restrictions for the AP600 methodology identified in Reference 3. The HOTSPOT computer code is used as described in WCAP-12945-P-A to calculate the AP1000 local oxidation. A WCOBRA/TRAC thermal-hydraulic transient which results in a nominal hot rod reflood phase PCT higher than the 95th percentile PCT is used in conjunction with a HOTSPOT computer code analysis in which time scaling is used to account for time at temperature.

The results of the AP1000 calculations are a maximum local oxidation of 12.9% and a core-wide maximum oxidation of 0.73%. The calculated values are in compliance with the 10CFR50.46 Acceptance Criteria values of 17% and 1%, respectively. Limitation 4d stated that the results of the additional calculations should be submitted for staff review. This is addressed through the presentation of the large break LOCA analysis results provided in AP1000 DCD section 15.6.5.4A (Revision 3).

B. The accumulator and CMT injection flow rates presented in Figures 15.6.5A-5 and -6 reflect the fact that after the RCS pressure has decreased to the accumulator setpoint, the flow delivered through the DVI lines is exclusively from the accumulators until their gas pressure has decreased to a value close to the RCS pressure. Both of the Core Makeup Tanks remain almost completely full until after 210 seconds of transient time have elapsed, when the intact loop CMT begins to inject once again. Both accumulators continue to inject until they empty more than 4 minutes into the transient. Since neither the accumulators nor the core makeup tanks empty during the time span of these figures, the integrals of the flow rates shown in Figures 15.6.5A-5 and –6 do not equal the initial contents of the respective tanks.

The graph requested is provided in the three figures attached. Three figures are used to show the requested flows and levels in different time segments to provide a reasonable resolution. The actuation of ADS-4 flow and IRWST injection is not modeled in the WCOBRA/TRAC run, but would occur due to a LO/LO level in the intact loop CMT between 1700-1800 seconds. The collapsed liquid level in the core is referenced to the bottom elevation of the active fuel , and the collapsed liquid level in the downcomer is referenced to the bottom of the reactor vessel. The Figures show that CMT injection is adequate to maintain mass inventory in the core and downcomer once the accumulator has emptied.

C. The cladding oxidation results are shown in Section A of this response.

The AP1000 is equipped with passive safety systems to mitigate postulated accidents, including LOCA events. For a design basis large break LOCA event, the only credible single failures that could affect the transient are (1) the failure of one of two parallel path valves to open in one of the two core makeup tank delivery lines, or (2) the failure of one of two parallel path valves to open in the PRHR return line. The CMT and PRHR are identified as of minor importance in the AP1000 large break LOCA PIRT (found in WCAP-14171, Rev.2). Reference 3 requires that the sensitivity of the PCT result to operation of each of these systems be established for the AP1000 by elimination of the system from the large break LOCA model. To comply with application limitation 4b, Westinghouse has performed runs to demonstrate that modeling the AP1000 CMT and PRHR systems is more conservative for calculated PCT than is ignoring their existence. The limitation 4b cases bound the impact of any single valve failure assumption in the system piping.

The single failure assumed in the AP1000 large break LOCA ECCS analysis presented in Chapter 15 of the DCD is the failure of a CMT delivery line isolation valve. This failure affects the safety injection flow delivery when the CMT begins to inject toward the end of accumulator injection. The large break LOCA case has been extended until 1800 seconds, a time at which the CMT liquid level has decreased to the LO/LO setpoint that activates both ADS stage 4 and the IRWST. The figures in Section B of this response show the core inlet flow, outlet flow, core collapsed liquid level and downcomer collapsed liquid level for the extended large break LOCA transient. The core collapsed liquid level in the figure is referenced to the bottom elevation of the active fuel , and the collapsed liquid level in the downcomer is referenced to the bottom of the reactor vessel.

D. The statement in the DCD about Figure 15.6.5A-12 is correct. The first figure from Part B illustrates the reason why the core and downcomer levels are approximately the same at 20 seconds transient time (50 seconds on Figure 440.097-1-1). At this time the downcomer is highly voided because its initial inventory has been discharged through the break and the accumulator injection is bypassing to the break. The source of the liquid in the core at this time is not flow from the downcomer but rather liquid that has entered the top of the core from the upper plenum as the upper head drains (shown as negative flow at the top of the core prior to 50 seconds on Figure 440.097-1) that proceeds through the core toward the lower plenum and the break location.

E. Table 15.6.5-4 of the AP1000 DCD indicates that the limiting hot assembly location was identified, consistent with the approved best estimate large break LOCA methodology, to be beneath an open hole in the upper core plate.

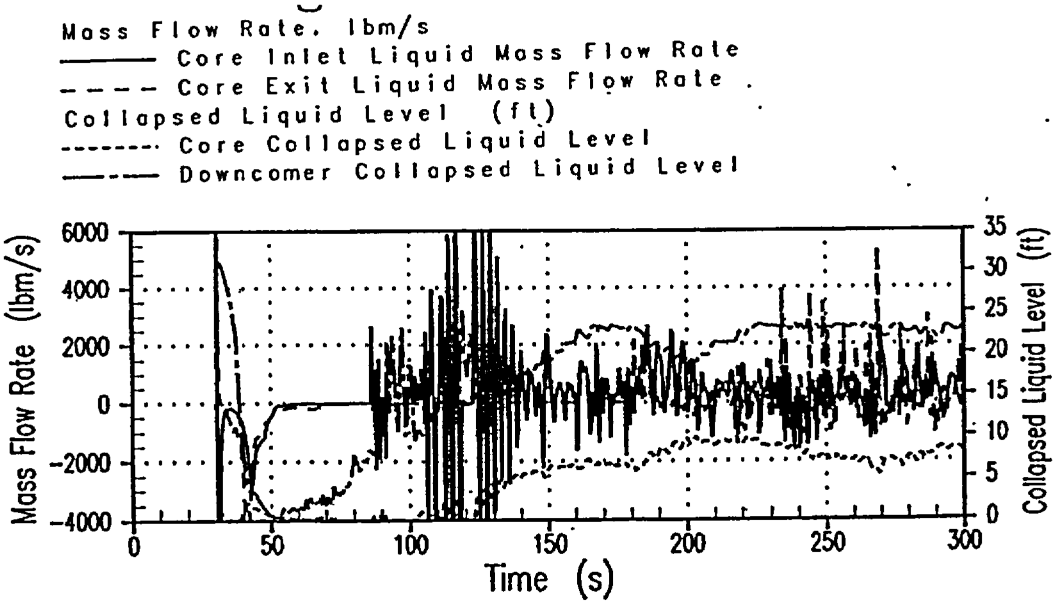


FIGURE 440.097 R1-1

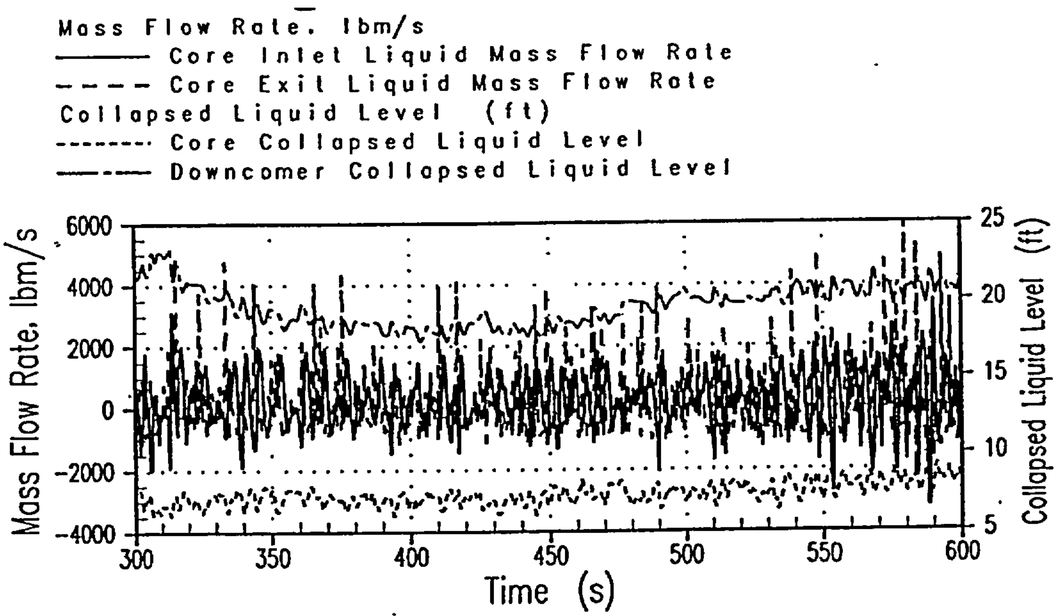


FIGURE 440.097 R1-2

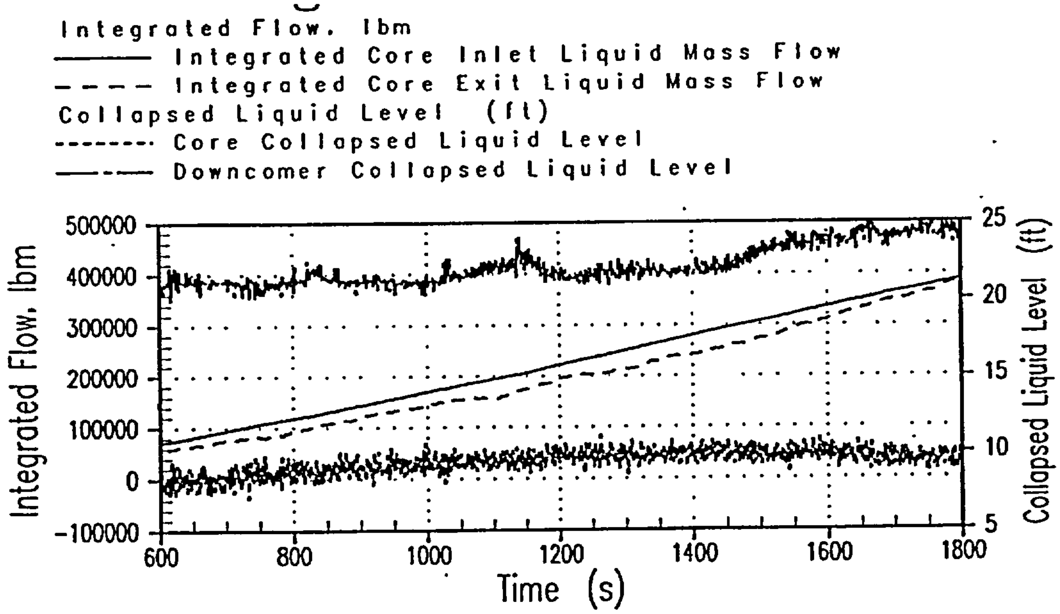


FIGURE 440.097 R1-3

Design Control Document (DCD) Revision:

DCD Chapter 15 (Revision 3) will be modified as shown in the attachment.

15.6.5.4A.5 Large-break LOCA Analysis Results

For the AP1000 large-break LOCA analysis, the best-estimate LOCA analysis methodology approved for AP600 is applied as follows. The plant boundary conditions for WCOBRA/TRAC, including the initial operating conditions and the core power distribution, are bounded in a conservative manner based on the sensitivity studies that investigated the range of AP600 possible values. Studies were reperformed for AP1000 to establish the bounding values for the AP1000 reference transient.

Conceptually, the following equation defines the effect on the reference transient PCT of the uncertainties due to global model parameter variations:

$$PCT_i = PCT_{REF,i} + \Delta PCT_{MOD,i}$$

where,

$PCT_{REF,i}$ = Reference transient PCT: The reference transient PCT is calculated using WCOBRA/TRAC at the bounding initial conditions and distribution for blowdown (i = 1) and reflood (i = 2).

$\Delta PCT_{MOD,i}$ = Model bias and uncertainty: This component accounts for uncertainties in the ability of the WCOBRA/TRAC code to accurately predict important phenomena affecting the overall system response ("global" parameters) and the local fuel rod response ("local" parameters). The code and model bias is the difference between the reference transient PCT, which assumes nominal values for the global and local parameters, and the average PCT, taking into account the possible values of global and local parameters. The global model matrix for AP1000 is presented in Reference 11.

Reference 3 indicates the application restrictions on the AP600 methodology. The AP1000 large-break LOCA analysis has complied with those restrictions. The global model matrix of calculations and the final 95-percent uncertainty calculations have been performed for AP1000. The reference transient was reanalyzed to address the sensitivity to the modeling of the CMT and PRHR. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the reference transient PCT. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the reference transient result. **Further, local and core-wide cladding oxidation values have been determined using the Reference 10 approved methodology.**

15.6.5.4A.8 Large-Break LOCA Conclusions

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

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

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

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

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

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

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

   Criterion 5 is satisfied if a coolable core geometry is maintained, and the core is cooled continuously following the LOCA. The AP1000 passive **core cooling** system provides effective core cooling **following a large-break LOCA event, even assuming the limiting single failure of a core makeup tank delivery line isolation valve. The large break LOCA transient has been extended beyond fuel rod quench until 1800 seconds, a time at which the CMT liquid level has decreased to the low-2 setpoint that actuates the 4th stage ADS valves and IRWST injection. A significant increase in safety injection flow rate occurs when the IRWST becomes active. The analysis performed demonstrates that CMT injection is sufficient to maintain the mass inventory in the core and downcomer, from the period of fuel rod quench until IRWST injection. The AP1000 passive core cooling system provides effective post-LOCA long-term core cooling.**

Table 15.6.5-8

BEST-ESTIMATE LARGE-BREAK LOCA RESULTS

Parameter	Value	Criteria
Calculated 50th percentile PCT (°F) (for time period of maximum 95th percentile)	1840	N/A
Calculated 95th percentile PCT (°F)	2124	2200
Maximum **local** cladding oxidation (%)	**< 12.9**	17
Maximum **core-wide cladding oxidation** (%)	**0.73**	1
Coolable geometry	Core remains coolable	Core remains coolable
Long-term cooling	Core remains cool in long term	Core remains cool in long term

PRA Revision:

None

RAI Number: 440.151 (Response Revision 1)

Question:

Section 2.2.1.5 states that the quality (*x*) entering the automatic depressurization system stage 4 (ADS-4) branch line from the hot leg is calculated using,

$$x = R^{3.25(1-R)^2} \tag{1}$$

where, $R = (h/h_b)$. Here, h is the distance between the top of the main pipe and the liquid surface, and h_b is the critical distance for entrainment onset. The onset of entrainment is obtained from an expression based on the gas phase Froude number (*Fr*) at the branch line inlet,

$$Fr_g = \frac{U_g}{\sqrt{\frac{gd\Delta\rho}{\rho_g}}} = C_1\left(\frac{h_b}{d}\right)^{C_2} \tag{2}$$

Various values of C_1 and C_2 have been proposed by different investigators, several of which are listed in Table 1.

Table 1

Reference	C_1	C_2
Anderson and Benedetti [1]	0.35	2.5
Rouse [2]	5.67	2.0
Schrock, et al. [3]	0.395	2.5
Smoglie [4]	0.355	2.5
Maciaszek and Micaelli [5]	1.75	1.5

Equations (1) and (2) are based on experiments in which the branch line diameter is small in comparison to the diameter of the horizontal pipe. In general, the ratio of the branch line to the main pipe was less than 0.1 in development of these correlations.

The coefficients used in WCOBRA/TRAC-AP are those of Anderson and Benedetti. The TRAC-M code also uses Equations (1) and (2) but with the coefficients by Smoglie. However, the coefficients by Smoglie and by Anderson and Benedetti are nearly identical, and the WCOBRA/TRAC-AP and TRAC-M models generate the same results. Figure 1 shows the predicted variation of branch line quality as a function of the branch line Froude number assuming the liquid level in the main pipe is at the midplane (h/D = 0.5) for two cases in which the ratio of the branch line to main pipe diameter is large compared to the database used to determine the coefficients in Table 1. ATLATS is a separate effects test facility with dimensions scaled to those of AP600 that was used to investigate phase separation at the junction between a small branch line and a main pipe. The ratio of the branch line to the main pipe in ATLATS is d/D = 0.33, which is less than the AP1000 ratio of d/D = 0.47. Calculations for both the ATLATS test facility geometry and the AP1000 hot leg and ADS-4 junction are also shown. As the ratio (d/D) increases, the model predicts lower branch line qualities. However, in neither case are the predicted branch line qualities in reasonable agreement with the experimental data shown in Figure 1. These data are from ATLATS tests, with the equilibrium level at approximately h/D = 0.5, consistent with the calculations.

The comparison suggests that the model and coefficients of Equations (1) and (2) and Table 1, grossly overpredict the ADS-4 quality for conditions expected in AP1000. Please provide suitable justification for Equations (1) and (2) and their coefficients for the large d/D ratios in the AP1000 design. Provide justification that the phase separation equations used in the WCOBRA/TRAC-AP code are appropriate for AP1000 ADS-4 analysis in light of these data.

References

[1] Anderson, J. L., and Benedetti, R. L., "Critical Flow Through Small Pipe Breaks," EPRI/NP-4532, 1986.

[2] Rouse, H., "Seven Exploratory Studies in Hydraulics," J. Hydr. Div. Proc. ASCE, HY4, pp (1038) 1-35, August, 1956.

[3] Schrock, V. E., Revankar, S. T., Mannheimer, R., and Wang, C-H., "Small Break Critical Discharge -- The Roles of Vapor and Liquid Entrainment in a Stratified Two-Phase Region Upstream of the Break," NUREG/CR-4791, 1986.

[4] Smoglie, C., "Two-Phase Flow Through Small Branches in a Horizontal Pipe with Stratified Flow, Ph. D. Dissertation, Univ. of Karlsruhe, 1984.

[5] Maciaszek, T., and Micaelli, J. C., "The CATHARE Phase Separation Model in Tee Junctions," SETh/LEML-EM/89-159, 1986.

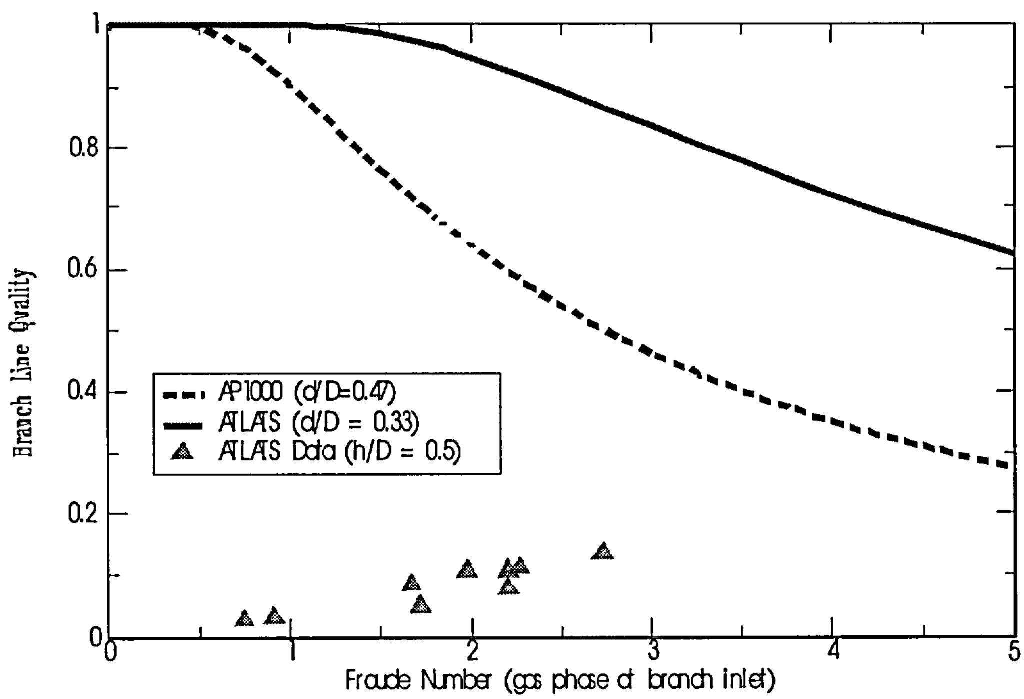


Figure 1. Branch line quality for liquid level at main pipe midplane for ATLATS and AP1000.

Westinghouse Response:

Entrainment of liquid from a stratified surface in a horizontal pipe to a vertical off-take has been studied and found by several investigators (including those cited in references 1 through 4 above) to correlate with Froude number and a geometric ratio of entrainment onset height to off-take diameter (Z/d). In section 4 of WCAP-15613 (AP1000 PIRT and Scaling Assessment) it was demonstrated that AP600 test facilities such as OSU adequately scale (relative to AP1000) Froude number in the hot legs and the resulting liquid entrainment inception correlation for stratified flow conditions. The entrainment correlations described in the references listed above use the same basic form consisting of Froude number and a geometric ratio of entrainment onset height to off-take diameter (Z/d). As the staff points out, there is some notable variation among the different experiments as different geometric ratios of off-take diameter to main pipe diameter (i.e. d/D) were tested and various coefficients (C1) or exponents (C2) were proposed to correlate the data as shown in Table 1 above. These investigations covered a range of d/D ratios from about 0.03 to 0.15. The d/D ratio for AP1000 is 0.47. Throughout the tested range, however, the investigations demonstrated that the basic form of the entrainment correlation remains valid as it appears to do a reasonable job of representing the experimental data. Therefore, although the AP1000 d/D ratio is outside the tested range, based upon the trend of these investigations, it is reasonable to expect the same form of the correlation to remain valid at larger d/D with only variation in the coefficient (C1) or exponent (C2) with scale. To address this, sensitivity calculations of entrainment from the hot leg to the ADS-4 offtake in AP1000 were performed with the WCOBRA/TRAC-AP models where the coefficient (C1) and exponent (C2) associated with entrainment inception correlation were varied as described in Appendix A.4 of WCAP-15833. The calculations indicated that the flow regime for AP1000 during the ADS-IRWST transition phase of a SBLOCA is predominantly stratified in the hot leg upstream of the ADS-4 off-take. Hence, the correlations are applicable. There is little sensitivity to variation in the coefficient (C1) or exponent (C2) associated with the entrainment inception correlation. Based upon the investigations at different geometric scales and the sensitivity study performed with WCOBRA/TRAC-AP it seems reasonable to apply these entrainment correlations to AP1000.

With respect to addressing the data associated with the ATLATS separate effects test facility, Westinghouse does not have detailed information regarding these tests (i.e facility layout, boundary conditions, test procedures, scaling analysis, etc.) and therefore these tests have not been analyzed by Westinghouse. Based on our understanding of the ATLATS test program, a possible explanation for the different behavior displayed in ATLATS (relative to the stratified flow type entrainment behavior expected in AP1000 and as seen in other test facilities) is that the ATLATS test facility is producing a different flow regime that could be attributed to its non-prototypic or incomplete simulation of the actual AP1000 configuration. The amount of effort necessary for Westinghouse to perform a detailed assessment of the ATLATS tests program, including the detailed evaluation of its design, scaling analysis, boundary conditions, test procedures, and test results that would otherwise be necessary for Westinghouse to consider the results of this test program are not warranted. Westinghouse believes that the staff can make the necessary safety determination for the AP1000 without a detailed assessment by Westinghouse of the ATLATS facility test results. We believe that WCAP-15833 Revision 1 provides the necessary information for the staff to determine the importance of upper plenum and hot leg entrainment and their effect on AP1000 plant safety.

NRC Additional Comments:

Notes: The response supplied in November 1, 2002 memo (W Ref.: DCP/NRC1529) is not sufficient. Westinghouse correctly notes in the their response that coefficients of entrainment onset correlations of the form:

$$Fr_g = \frac{U_g}{\sqrt{\frac{gd\Delta\rho}{\rho_g}}} = C_1\left(\frac{h_b}{d}\right)^{C2}$$

have been determined from experiments in which the value of d/D is significantly smaller than that in AP1000. In addition, Westinghouse points out on pages 2-3 and 2-4 of WCAP-15833 that the general form of this correlation is questionable. It lacks a dependence on viscous effects and surface tension, which have been found in other studies to be important in predicting the onset of entrainment. Thus, Westinghouse has not provided justification that the above relation is “reasonable” when it is applied well outside of its established range of validity.

In its response, Westinghouse cites sensitivity studies using WCOBRA/TRAC-AP in which coefficients of the entrainment onset correlation were varied and makes the claim that the correlation is applicable because the sensitivity is small. Westinghouse has not claimed, nor attempted to demonstrate that the coefficients were varied over a sufficiently wide range. In the Figure above, it is clearly evident that representative data from the ATLATS has a much lower flow quality to the ADS-4 than would be predicted by traditional entrainment rate correlations. For the Westinghouse sensitivity study to have value, the coefficients should be ranged such that predicted flow quality is approximately that of the experimental data. Then, if the delay in IRWST initiation (as in the comparisons of FIgures A.4.4-3, A.4.4-6, and A.4.4-9) remains small, then it will be demonstrated that AP1000 is not sensitive to hot leg entrainment.

In the final paragraph of the response suggests that flow regime in hot leg of the ATLATS tests is not the same as in AP1000. Westinghouse fails to recognize that the hot leg flow regime is indeed an important factor in phase separation at the ADS-4 branch line. What sets AP1000, APEX, and ATLATS apart from most other geometries in which phase separation to an upward oriented branch line occurs, is that the steam generator (SG) inlet plenum acts to trap water. In studies such as those by Shrock [3], the pipe exit in unrestricted, and a horizontal stratified flow pattern was established. In ATLATS, visualization showed that the flow pattern in the hot leg was not horizontal stratified. The SG inlet plenum caused oscillating plugs to form in the region between the branch line and the plenum. High rates of entrainment occurred when the liquid plug periodically covered the branch line inlet. Clearly, the mechanisms responsible for phase separation to the branch line are different in the case of ATLATS than in conventional studies such as Shrock.

Westinghouse may not be aware that the ATLATS facility is scaled 1:1 with APEX and tests were run so that gas velocities in the hot leg and branch line were comparable. Thus, the flow patterns that existed in ATLATS should also be present in APEX tests. Assuming APEX is correctly scaled, the same flow patterns should be expected in the AP600 and AP1000.

To resolve this RAI, the following approach is suggested:

(1) Augment the WCOBRA/TRAC-AP sensitivity study on hot leg entrainment onset by providing an additional case in which the models are biased to predict the very low branch line flow qualities suggested by the ATLATS data. The Westinghouse case would be acceptable if the delay in IRWST remains small.

(2) The scaling rationale presented on pages 4-38 and 4-39 of WCAP-15613 should be revised to make use of the onset correlation by Welter, K. B., Wu, Q., Yao, Y., and Reyes, J. N., "New Model for Predicting Two-Phase Entrainment Rates in Tees," ANS Trans., Winter Annual Meeting, Washington DC, 2002. Their onset correlation, which is based on ATLATS data and thus presumably embeds the effect of hot leg flow pattern, is given by:

$$(w_3^2)^* = \frac{K\left(\frac{h_b}{d}\right)^3\left[a\left(\frac{h_b}{d}\right)+1\right]^2}{\left[1-\left(\frac{h_b}{D}\right)^2\right]}$$

where,

$$(w_3^2)^* = \frac{w_3^2}{d^5 \rho_g g \Delta\rho}$$

In these expressions, w_3 is the mass flow rate in the branch line, ρ is the density, d is the branch line diameter, D is the main pipe diameter, h_b is the entrainment height, and K=0.66 and a=0.22 are experimentally determined constants.

A stronger argument that the APEX tests were appropriately scaled for hot leg entrainment might be made by using Equation (3). By using the definition in Equation (4), and assuming pressure similitude,

$$\left\{\frac{\dot{q}_{core}^{2}\left[1-\left(\frac{h_b}{D}\right)^2\right]}{d^5\left(\frac{h_b}{d}\right)^3\left[a\left(\frac{h_b}{d}\right)+1\right]^2}\right\}_R = 1.0$$

Using the appropriate values for pipe diameters, values for Π_R in the following Table can be generated for various assumptions on entrainment height

h_b	Π_R
0.05 * D	1.97
0.50 * D	1.71
0.95 * D	1.55

For each height, the scaling ratio is within the acceptance range of 0.5 < □R < 2.

Westinghouse Additional Response:

WCAP-15833, Rev. 2 provides a sensitivity study to the hot leg entrainment onset in Section A.4.4. Three cases are presented in which either coefficient C1 or C2 in the Froude number expression shown above in the additional comment section is varied from its base value. Within the WCOBRA/TRAC computer code, when either coefficient is varied the predicted branch line quality is also varied according to Equation(1) in the RAI statement.

The two coefficients can be manipulated together to obtain a very low branch line quality comparable to that of the Figure 1 ATLATS test data. Figure 2 (shown in the RAI 440.154 RAI statement) plots the ATLATS branch line quality data as a function of Froude number together with the AP1000 values calculated for the C1 and C2 values analyzed in Section A.4.4 of WCAP-15833, Rev. 2. Branch line qualities in the range of the ATLATS data can be obtained by further adjustments of the C1 and C2 coefficients. Specifically, at a Froude number of 2.0, applying C1 and C2 values of 0.10 and 2.0, respectively, predicts a branch line quality of 0.065 in RAI 440.154, Figure 2. The impact of these coefficient values on predicted behavior during the AP1000 ADS-4 IRWST initiation phase has been determined by performing another WCOBRA/TRAC sensitivity run for the Inadvertent ADS actuation scenario. The results are compared with the base hot leg entrainment onset case in the attached figures. The net result of the sensitivity case is the release of more mass through the ADS-4 flow paths early in the depressurization at a higher pressure, which results in less mass present in the upper plenum and hot legs at lower pressures. During the initial portion of the depressurization transient, added mass is released through the ADS-4 flow paths, and the reactor vessel and upper plenum mass is lower than in the base case, as shown in Figure 440.151-1-1. The lower mass present in the upper plenum leads in turn to a lower flow of entrained liquid into the hot legs beyond 100 seconds, so more effective depressurization occurs (Figure 440.151-1-2). The venting of steam as the pressure approaches IRWST actuation pressure through the ADS-4 paths is more effective, and IRWST injection begins slightly earlier in the sensitivity case, as shown in Figure 440.151-1-3.

In summary, this sensitivity case confirms the conclusion of the Section A.4.4 analysis that the AP1000 ADS-4 IRWST initiation phase behavior is insensitive to the hot leg entrainment onset prediction. Hot leg entrainment is therefore not a safety-significant parameter for the AP1000 performance during the ADS-4 IRWST initiation phase.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None

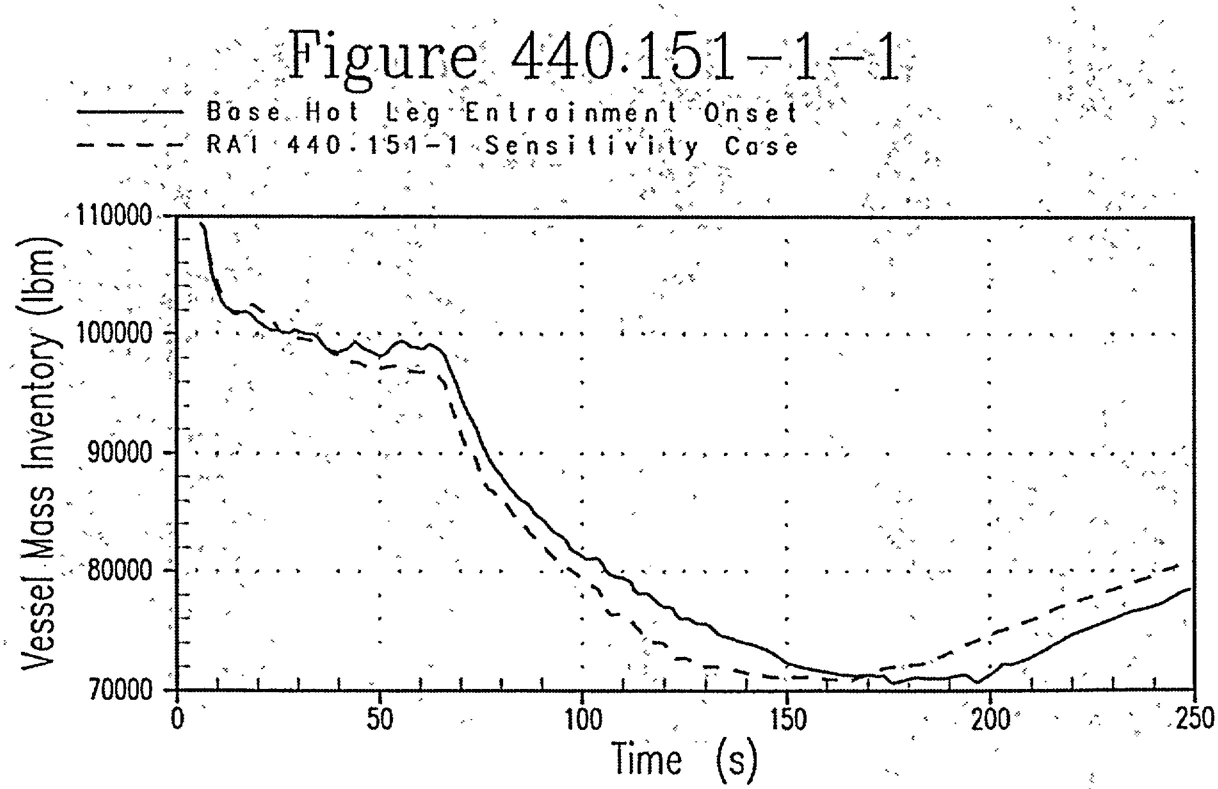
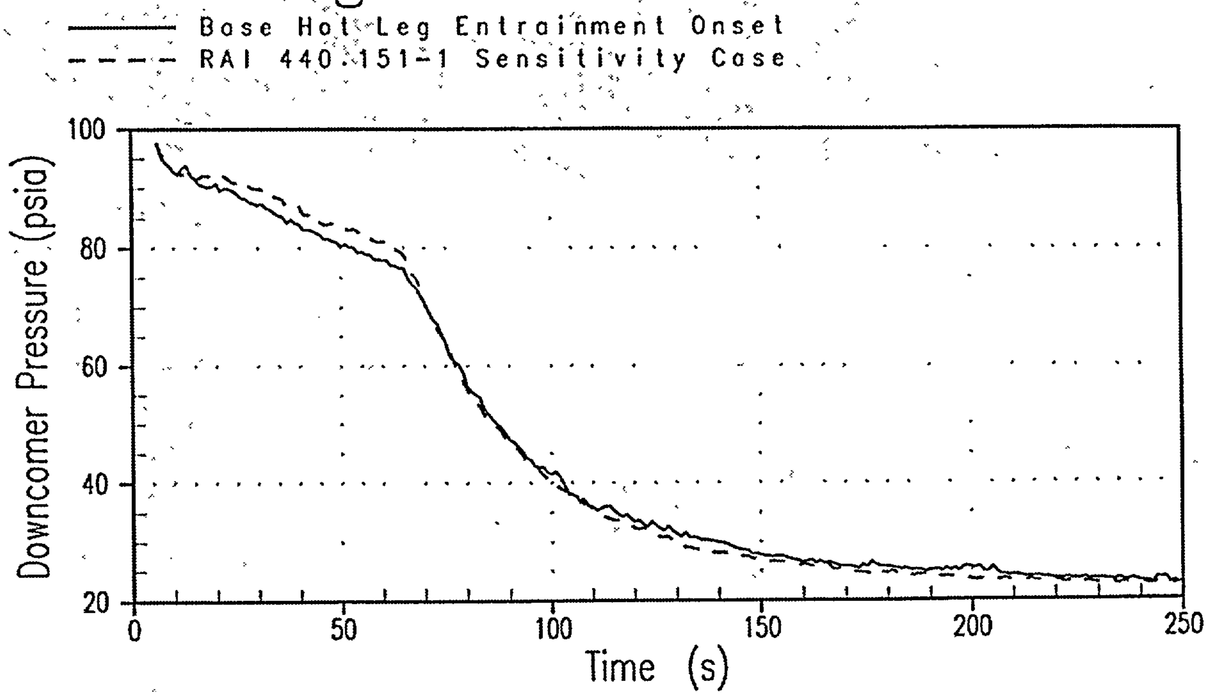


Figure 440.151-1-1
Base Hot Leg Entrainment Onset
RAI 440.151-1 Sensitivity Case
Vessel Mass Inventory (lbm)
110000
100000
90000
80000
70000
0
50
100
150
200
250
Time (s)

Figure 440.151-1-2



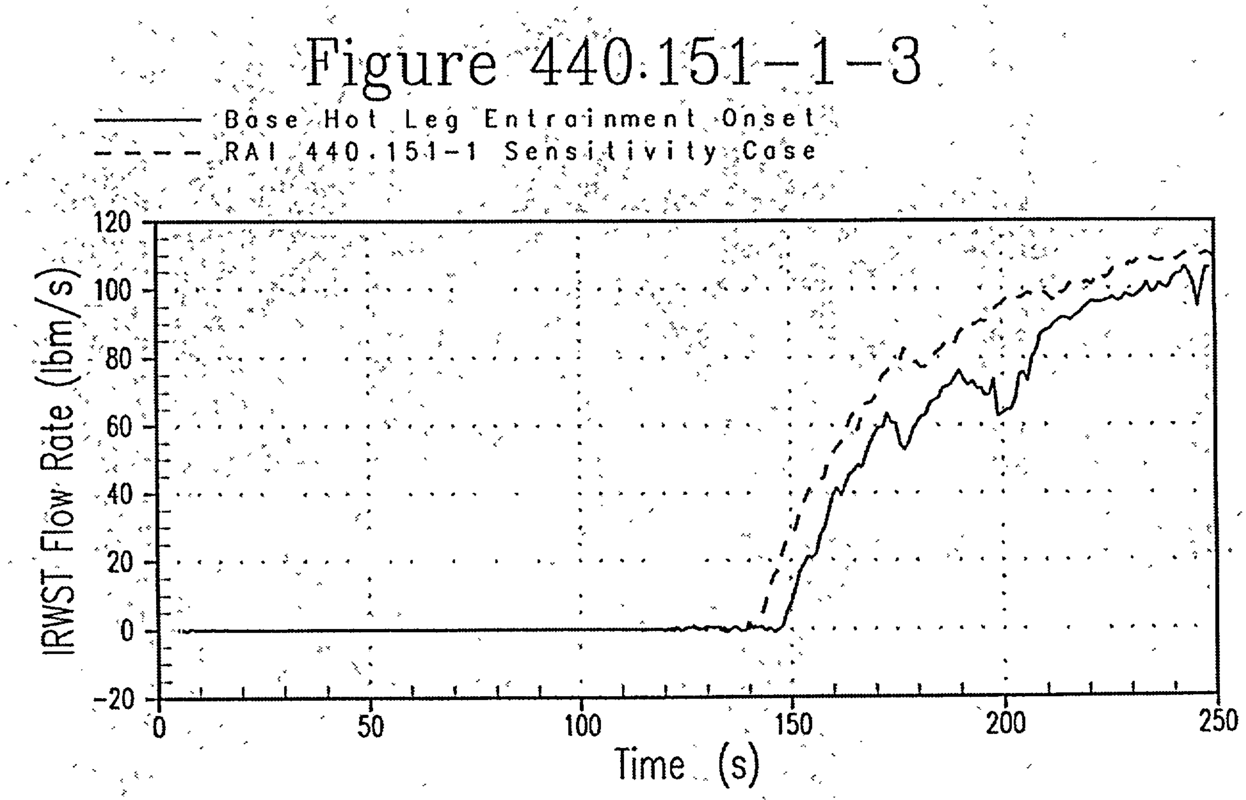


Figure 440.151-1-3
Base Hot Leg Entrainment Onset
RAI 440.151-1 Sensitivity Case
IRWST Flow Rate (lbm/s)
120
100
80
60
40
20
0
-20
0
50
100
150
200
250
Time (s)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.152 (Response Revision 1)

Question:

In Section 2.2.1.5 application of Entrainment/Vapor Pull-through Model is described. Under the "Model as Coded" subsection Step 6 is a branch line void fraction calculation. Please describe the slip or other models at the branch line junction to obtain a void - quality relationship. Since WCOBRA/TRAC-AP also calculates entrainment from a horizontal stratified flow using the models described in Section 2.2.1.4, describe how the liquid flow rate at the junction is determined from the entrained and continuous liquid fields.

Westinghouse Response:

The available correlations specify the branch line quality as a function of main line liquid level, and the level at onset of entrainment. The branch line quality then is defined as,

$$X_{BR} = \frac{\rho_G V_G \alpha_{BR}}{\rho_G V_G \alpha_{BR} + \rho_L V_L (1 - \alpha_{BR})}$$

Solving for α_{BR} as,

$$\alpha_{BR} = \frac{\rho_L V_L X_{BR}}{\rho_G V_G (1 - X_{BR}) + \rho_L V_L X_{BR}}$$

Where X_{BR} is the branch line quality, α_{BR} is the branch line void fraction, and V_G and V_L are phasic velocities. In the actual coding these velocities were taken from the previous time step.

In the current model, the collapsed liquid level is used for the prediction of branch line quality. Therefore, the entrained droplets which may be present in the stratified flow, will be added to the actual level in the main line.

If the predicted branch line void fraction is less than the donor cell void fraction (in this application, the top cell of the main line channel). The branch line void fraction is set to the void fraction in the top cell of the main line channel where the 1D component is attached.

NRC Additional Comments:

Response supplied is not sufficient. Additional information is needed to describe what the code does when the quality predicted by an offtake expression does not correspond to the quality (inferred from the void fraction) in the top cell of the Channel in the hot leg where the branch line connects. For example, if there are insufficient droplets in the top cell to satisfy the branch line quality demand, where does the "extra" liquid get extracted from ? Are mass and momentum conserved as implemented?

Westinghouse Additional Response:

HORIZONTAL STRATION ENTRAINMENT (HSE) MODEL AT BREAK/BRANCH

MODEL DESCRIPTION

The HSE model is in effect when:

- User option identifies a particular 1D/3D junction to enable the HSE model.
- Main pipe is horizontally stratified according to Taitel-Dukler Horizonal Flow Regime Map.
- Froude number in excess of onset criteria.

Overview of the model

If above conditions are not met, the junction behaves as a regular 1D/3D junction and convect the contents of a vessel fluid cell connected to the 1D/3D junction.

[

]a,c

a,c

a, c

a, c

MASS AND ENERGY CONVECTION

Pertinent Terms appearing in Vessel Field Equations

[

a,c

]

Pertinent Terms in 1D Mass and Energy Equations

For 1D junction mass and energy balance, the weighted average scalar variables are computed when the flow is from 3D to 1D and the convection from three levels in the vessel are summed up and used as the convected term in 1D mass and energy equation.

MOMENTUM CONVECTION

Vessel Equations

For HSE multi-cell connection, the general scheme is followed to account for the momentum balance using the junction phasic velocities in the VESSEL. Figure below shows the convected terms in a regular 1D/3D junction momentum.

a, c

1D Mixture Momentum Equation

For 1D junction momentum balance, the momentum convection for three levels in the vessel are summed up and used as the momentum convection term in its mixture momentum equation so that the 1D junction network equations are not altered from the original (1 level 1D/3D connection).

WHEN MAIN PIPE IS NOT STRATIFIED

When the main pipe is not stratified, the weighting factor is set to 0.0001, 0.0001, 0.9998 to bottom, middle, and top respectively so that only the content of top cell is allowed to convect to 1D PIPE. This is equivalent to the regular 1D/3D connection.

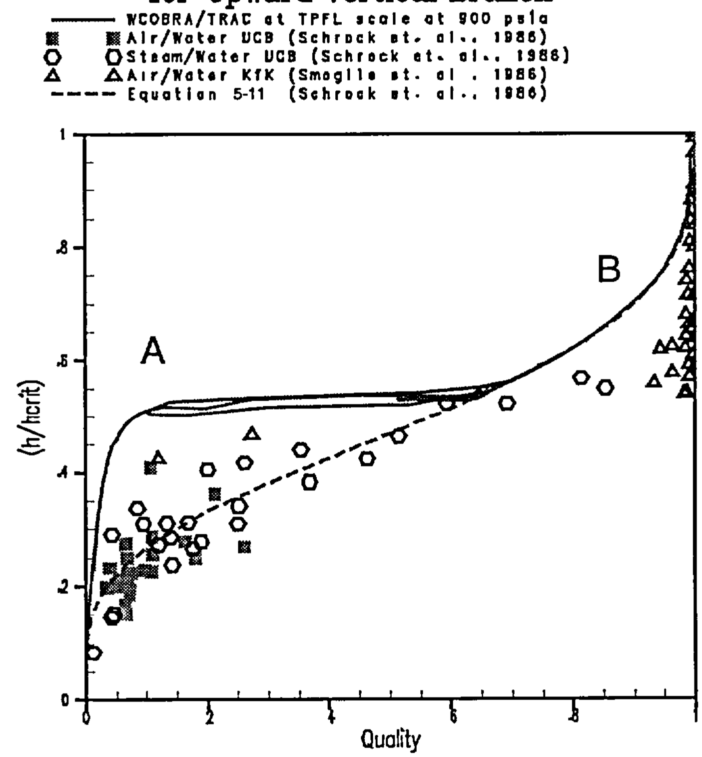
PERFORMANCE OF HSE MODEL FOR TOP VERTICAL CONNECTION

Test data with a branch-line connection at the top of the main pipe was not readily available (TPFL [3] did not examine this configuration). Therefore, a correlation by Schrock [6] (implemented into Relap5 by Ardron and Bryce [4]) and normalized data points from other experiments [5], [6] were used to compare the code prediction.

In the figure below, h is the distance between the break elevation and the stratified surface. Hcrit is the distance at which the entrainment begins. The comparison plot indicates that more liquid is predicted to entrain when the liquid level is close to the top of main pipe than was observed in the experiment and also the correlation (Point A).

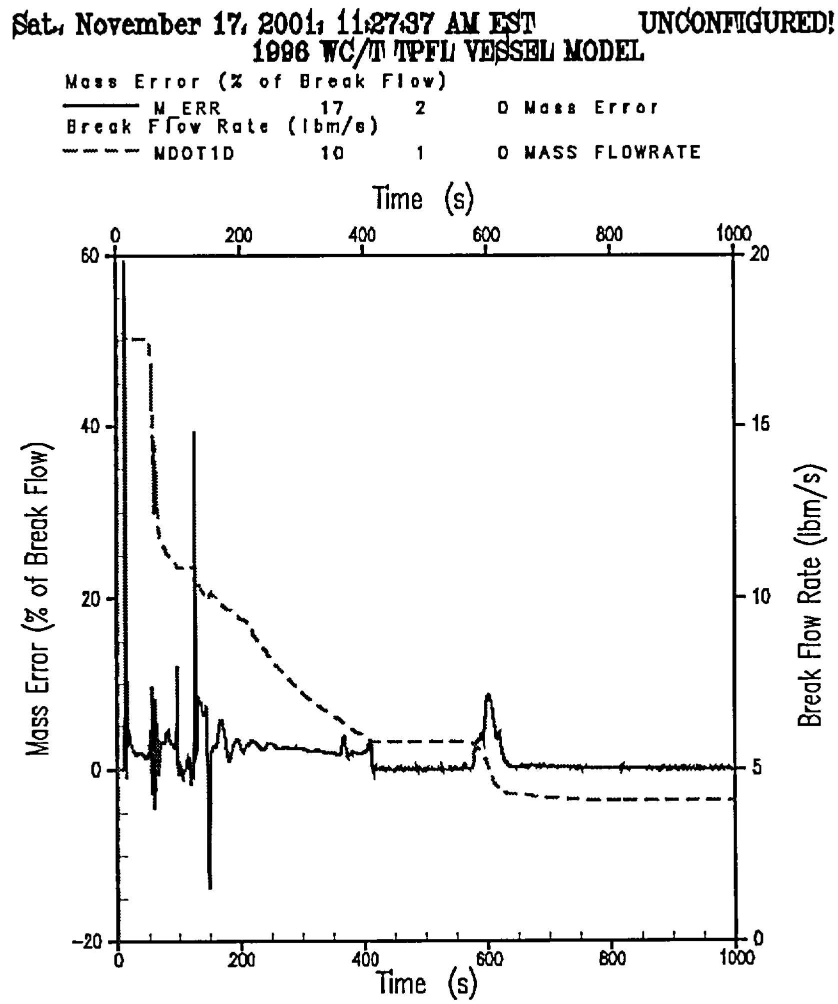
This behavior is expected because the donor void fraction is always bounded by the cell void fraction in the channel connected to the 1D component. The Ardron and Bryce correlation tends to entrain more liquid when the liquid level becomes closer to the critical height (Point B).

WCOBRA/TRAC Entrainment Prediction vs. Data for Upward Vertical Branch



MASS BALANCE

Mass balance is checked in one of TPFL test with the horizontal connection in the figure below. The mass balance across the 1D/3D connection was checked by taking the difference between the summation of adjacent gap flows and the 1D component break flow. This ignores the storage term in the volume of the channel connected to the 1D. By ignoring the storage term, the apparent mass balance is contaminated when the junction cells are either depleting or accumulating liquid, as in the time periods between 0-400 seconds, and 600-650 seconds in the attached figure. In the quasi-steady state time periods, the mass balance error is essentially zero.



MOMENTUM BALANCE

Single Level 1D/3D Connection Check:

Figure below shows the typical pressure drop prediction using the original 1D/3D (single level 1D/3D connection).

To examine the momentum convection scheme, a simple vertical channel/pipe model with FA=1ft^2, ΔX=1ft was constructed as seen in the left figure.

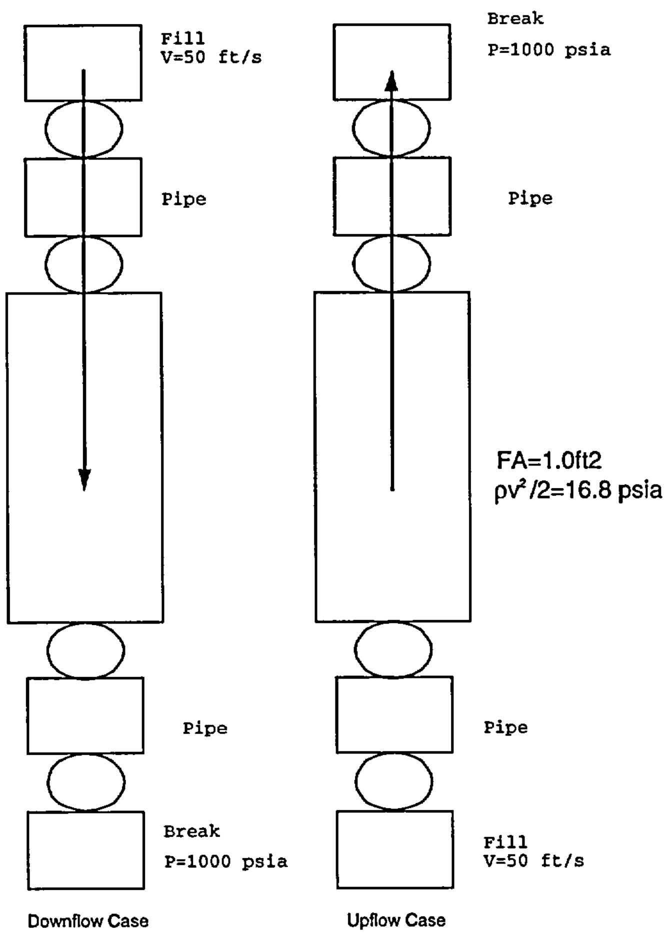


Figure below depicts the static pressure field in this simple geometry. In this problem, each cell is 1 ft high, the flow area is 1 ft^2. The break outlet boundary condition is set to 1000 psia, the fill inlet boundary condition is set to 50 ft/sec. The gravitational head is 0.43 psia/ft, the frictional pressure drop is 0.08 psia/ft. At ρ=62.2(lbm/ft^3) and v=50 ft/s, $\rho v^*v=\rho v^2$=62.2*2500 (lbm/ft^3)*(ft^2/s^2)=33.54 psia.

Pressure Field at Pipe-Vessel-Pipe is shown below:

```
============================
                                    Time is   10.0 second

             50.03 ft/s          -50.00 ft/s
           -----------         -----------
           | 1000.54|          |  998.53|      ----- 1D Cell
          -------------       -------------
          |  1001.12 |        |   998.79 |
          -------------       -------------
          |  1001.66 |        |   999.15 |
          -------------       -------------        Vessel Cells
          |  1002.19 |        |   999.48 |
          -------------       -------------
          |  1002.69 |        |   999.81 |
          -------------       -------------
           | 1003.30|          | 1000.09|      ----- 1D Cell
           -----------         -----------

              Upflow              Downflow
```

The pressure drop across each of 1D and 3D cells are approximately equal to ~0.51 psia, which is the gravitational head plus the friction loss for the upflow case, and ~0.35 psia, which is the gravitational head minus the friction loss for the downflow case. In these simple 1D/3D connections, a reasonable pressure drop is predicted.

HSE's Multi-level 1D/3D Connection Check:

In a complicated geometry involving a 90º bend such as one at the 1D/3D junction at ADS4 connection requires an application of the turning loss associated with the 90º bend and/or the entrance loss into the branchline in addition to K=1 for preserving the dynamic head at the 1D cell face at the junction, as described in Section 4-7-4 of CQD [1].

CONCLUSION

The HSE model implemented in WCOBRA/TRAC-APSB version accounts for all important conservation equations and the model's approximation is adequate for the intended application.

REFERENCES

[1] S. M. Bajorek, et al, "Code Qualification Document," Volume 1, WCAP-12945-P-A.

[2] W. L. Brown, et al, "WCOBRA/TRAC AP1000 ADS-4/IRWST Phase Modeling," WCAP- 15833-P Rev. 2.

[3] Anderson J. L. and Benedetti, R. L. "Critical Flow through small pipe break," EPRI NP- 4532, May, 1986.

[4] K. H. Ardron and W. M. Bryce, "Assessment of horizontal stratification entrainment model in RELAP5/MOD2 by comparison with separate effects experiments", Nuclear Engineering and Design 122 (1990) 263-271.

[5] Schrock, V. E. and Revankar, S. T., Manheiner, R. and Wang, C. H. "Small break critical discharge - the roles of vapor and liquid entrainment in a stratified two-phase region upstream of the break," NUREG/CR-4761, December 1986.

[6] Smoglie, C. and Reimann, J. "TWO-PHASE FLOW THROUGH SMALL BRANCHES IN A HORIZONTAL PIPE WITH STRATIFIED FLOW," International Journal of Multi-Phase Flow pages 609-626, 1986.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.002 (Response Revision 1)

Question:

Please provide the following information with regard to the radiological consequences analysis of the design-basis Locked Rotor Accident (LRA) as discussed in Chapter 15.3.3 and Table 15.3-3 of the AP1000 DCD:

A. It is stated that it was determined that as a result of the LRA no fuel is damaged such that the activity in the fuel-cladding gap is released, but that a conservative assumption of 16% of the core fuel rods failed was used in the radiological consequences analysis. How was it determined that no fuel is damaged? What is the basis for the assumption of 16% failed fuel?

B. What is the basis for the assumed accident duration of 1.5 hours for the LRA?

C. What assumptions were made in the determination of the steam mass release from the secondary system associated the radiological consequences analysis of the LRA?

D. What is the basis for the leak flashing fraction of 0.04% for the first 60 minutes of the LRA?

E. Table 15.3-3 lists the reactor coolant noble gas activity as equal to the operating limit of 280 milliCi/gm (milli-Curies-per-gram) dose equivalent Xe-133. Other accidents list this operating limit as 280 microCi/gm dose equivalent Xe-133. Please clarify the discrepancy (is this a typographical error)?

F. Table 15.3-3 lists a fission product gap fraction of 0.10 for Kr-84. The krypton isotope of concern with respect to gap fractions for non-LOCA design-basis accident dose analyses is Kr-85. Please clarify the correct isotope of Kr (is this a typographical error)?

Westinghouse Response:

A. The response to RAI Number 440.080 discusses the basis for the determination that no fuel is damaged as a result of the design basis locked rotor accident. The bounding dose analysis is performed assuming some fuel failure and 16 percent was selected based on preliminary conservative fuel failure assessments.

B. Safety-related decay heat removal is provided by the Passive Residual Heat Removal (PRHR) System. In the event of a locked rotor there may be no primary system signal that actuates the PRHR. In this case decay heat is removed by steaming off the secondary inventory until the low steam generator level signal actuates the PRHR. Eventually the PRHR system is removing all decay heat and heat transfer to the steam generators stops, thus terminating steaming. It was determined that this would occur within the first 1.5 hours.

   If startup feedwater is available, then steam releases would continue until the normal residual heat removal system (RNS) is operating and removing all decay heat, which occurs within eight hours of event initiation. This was determined to be a less limiting scenario since, with startup feedwater available, the iodine and alkali metal activity contained in the primary to secondary leakage that flashes is not directly released but is assumed to mix in the secondary liquid and be released with the steam, subject to partitioning.

C. As discussed above, steam releases stop when the PRHR is removing all decay heat. This occurs within the first 1.5 hours for the case with no startup feedwater available. The steam releases were calculated assuming the maximum initial steam generator water inventory. This maximizes both the time until the PRHR setpoint is reached and the mass of steam that is released until that time. The analysis also modeled minimum PRHR heat transfer capability.

D. The temperature of the hot leg following the locked rotor was used together with the secondary pressure to calculate the flashing fraction. Following PRHR actuation the primary temperature drops to the extent that the leak flow no longer flashes. This occurs before 1 hour after event initiation. The average flashing fraction over the hour was calculated to be less than 0.04 (i.e., less than 4% of the leakage flashes). Table 15.3-3 is being corrected to show the flashing as a fraction (0.04) with no units.

E. The reactor coolant noble gas activity used in the locked rotor analysis was 280 μCi/gm, dose equivalent Xe-133. The typographical error is being corrected.

F. The locked rotor analysis modeled Kr-85 with a gap fraction of 0.10. Kr-84 (which is stable) was not modeled. The typographical error is being corrected.

Design Control Document (DCD) Revision:

From DCD Chapter 15.3,Table 15.3-3:

Table 15.3-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/gm of dose equivalent I-131 (see Appendix 15A)[(a)]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/gm dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of design basis reactor coolant concentrations at maximum equilibrium conditions
Fraction of fuel rods assumed to fail	0.16
Core activity	See Table 15A-3
Fission product gap fractions	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Reactor coolant mass (lb)	3.7 E+05
Secondary coolant mass (lb)	6.06 E+05
Condenser	Not available
Duration of accident (hr)	1.5 hr
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	350[(b)]
Steam released (lb)	
0-1.5 hours[(c)]	6.48 E+05
Partition coefficient in steam generators for iodine and alkali metals	0.01
Leak flashing fraction[(d)]	
0-60 minutes	0.04
> 60 minutes	0

PRA Revision:

None

NRC Additional Comment:

D. What is the basis for the leak flashing fraction of 0.04% for the first 60 minutes of the LRA?

The Westinghouse response clarified that the leak flashing fraction is 0.04 (no percent). The response stated that the flashing fraction was calculated using the temperature of the hot leg following the locked rotor together with the secondary system pressure. What is the temperature of the hot leg? Is this noted somewhere in the design control document (DCD)?

Westinghouse Additional Response:

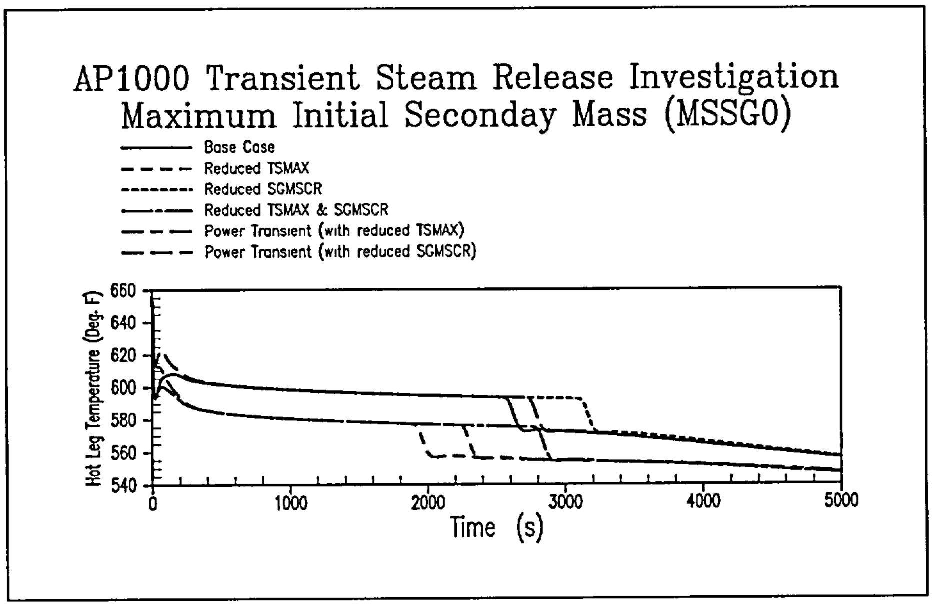
The reactor coolant temperatures following the locked rotor were used together with the secondary pressure to calculate the flashing fraction. This flash fraction was determined on a time dependent basis using a constant enthalpy process based on enthalpy of the primary coolant in the steam generator tubes, the enthalpy of liquid water at saturation, the time dependent enthalpy conditions at the secondary side pressure, and the heat of vaporization at the secondary side pressure. It is generally assumed that the primary to secondary leakage is distributed throughout the tube bundle and a flashing fraction based on the average temperature of the hot and cold legs is justified. However, since this is a critical parameter for the dose analyses, and the actual location of the leak(s) is unknown, a conservative flashing fraction skewed towards the hot leg temperature was used.

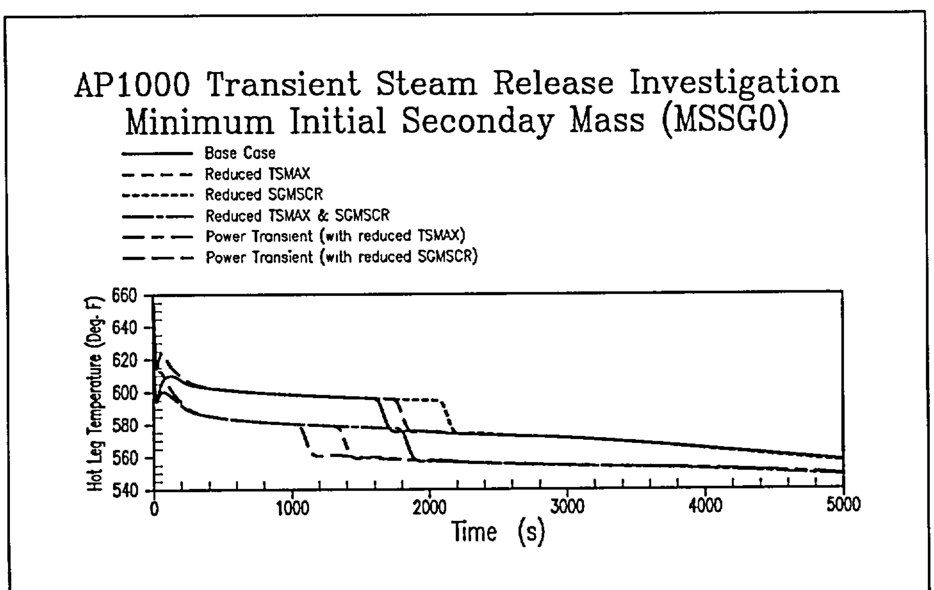
The LOFTRAN model was used to calculate the transient reactor coolant system temperatures and secondary side pressures following a postulated locked rotor event. The transient models the reactor trip with subsequent loss of offsite power. Main feedwater is lost and no startup feedwater is provided. Decay heat removal occurs through the steam generators. As secondary mass is lost due to steaming, PRHR actuation occurs on low steam generator level. Eventually the PRHR is removing all decay heat and heat transfer to the steam generators, and steam release stops. Sensitivity cases were run to investigate the impact of parameters which potentially impact the results being investigated, including: the initial secondary side mass, the steam generator safety valve setpoint, the PRHR actuation setpoint, and an initial power excursion. Bounding values were developed based on the results.

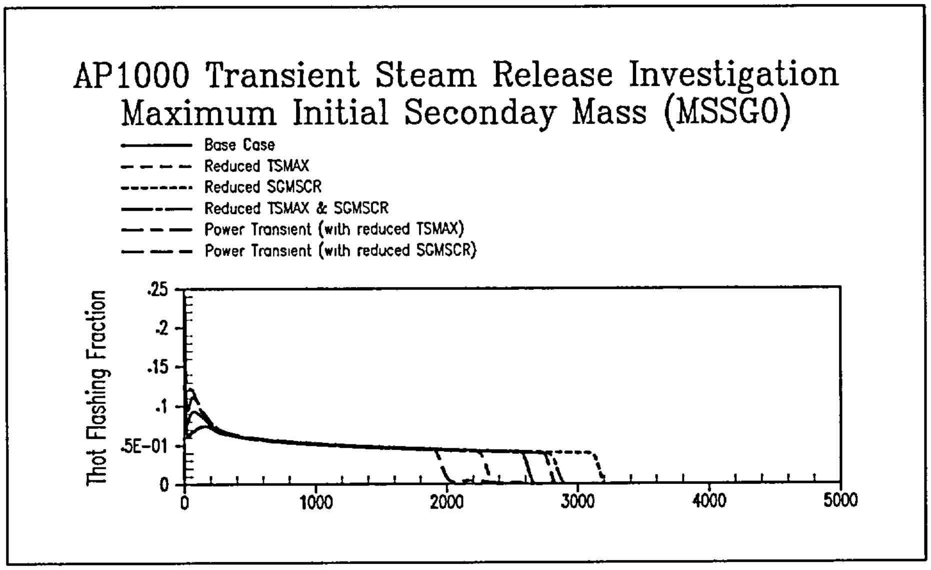
The time dependant flashing fraction was calculated assuming all leakage was at the temperature of the hot leg and a bounding average flashing fraction of 0.055 was established. The calculation was repeated assuming that all leakage was at the average of the hot and cold leg temperatures, and a bounding average flashing fraction of 0.025 was established. For the dose analysis these two values were averaged to a flashing fraction of 0.04.

Following PRHR actuation the primary temperature drops to the extent that the leak flow no longer flashes. The determination of the time that flashing of primary to secondary leakage stops was conservatively determined based on the assumption that all leakage is at the hot leg temperature.

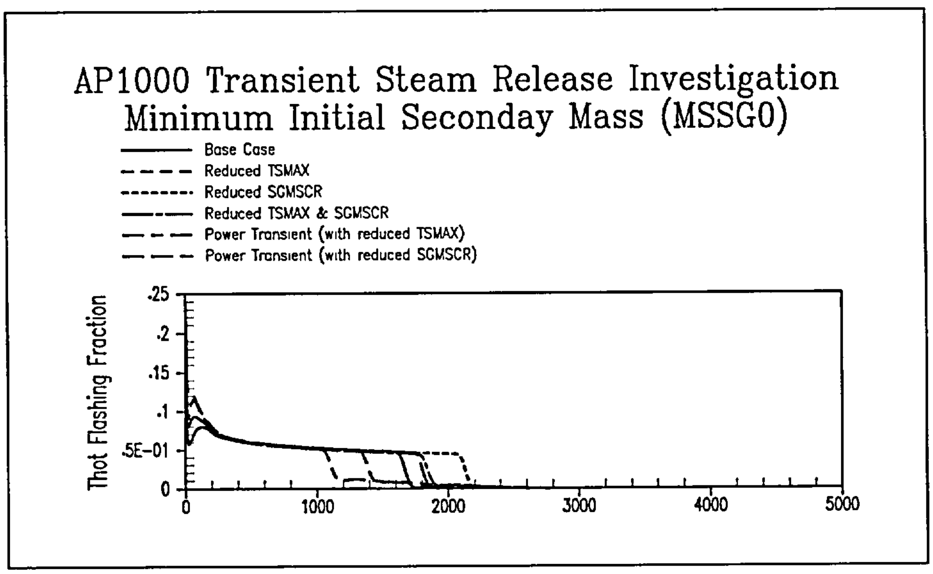
The figures below present some of the transient data considered in determining the flashing fraction for use in the locked rotor dose analysis. This background information is not typically provided in the DCD.



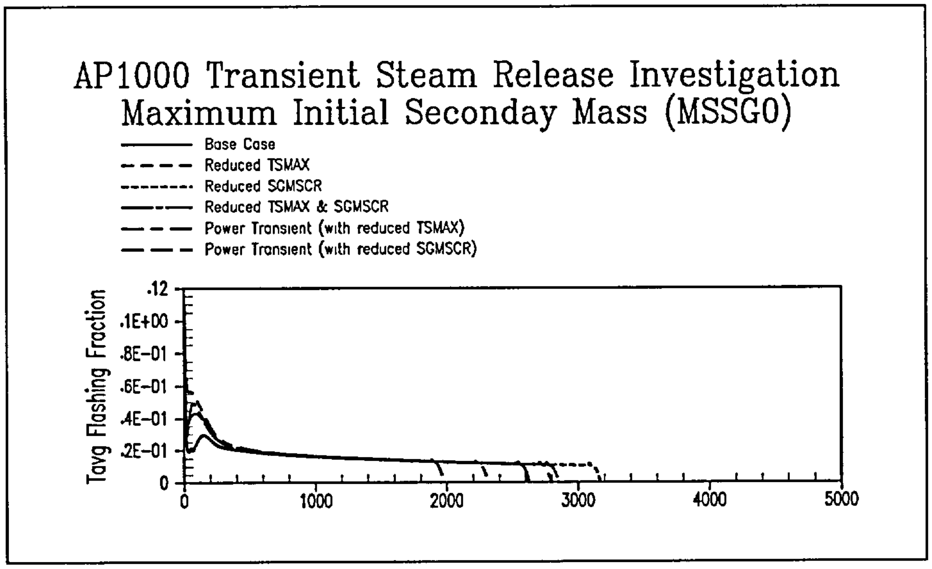




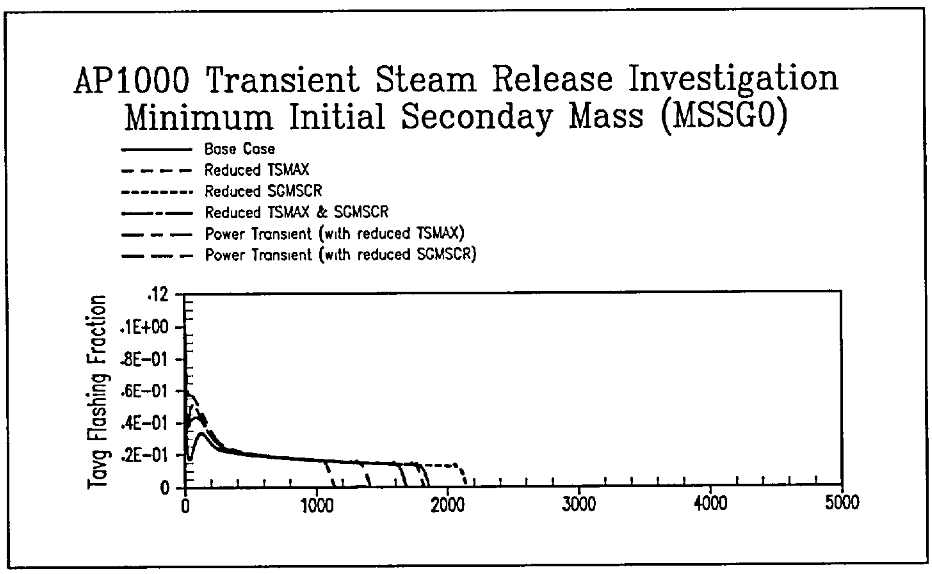
AP1000 Transient Steam Release Investigation
Maximum Initial Seconday Mass (MSSG0)
Base Case
Reduced TSMAX
Reduced SGMSCR
Reduced TSMAX & SGMSCR
Power Transient (with reduced TSMAX)
Power Transient (with reduced SGMSCR)
Thot Flashing Fraction
.25
.2
.15
.1
.5E-01
0
0
1000
2000
3000
4000
5000



AP1000 Transient Steam Release Investigation
Minimum Initial Seconday Mass (MSSG0)
Base Case
Reduced TSMAX
Reduced SGMSCR
Reduced TSMAX & SGMSCR
Power Transient (with reduced TSMAX)
Power Transient (with reduced SGMSCR)
Thot Flashing Fraction
.25
.2
.15
.1
.5E-01
0
0
1000
2000
3000
4000
5000



AP1000 Transient Steam Release Investigation
Maximum Initial Seconday Mass (MSSG0)
Base Case
Reduced TSMAX
Reduced SGMSCR
Reduced TSMAX & SGMSCR
Power Transient (with reduced TSMAX)
Power Transient (with reduced SGMSCR)
Tavg Flashing Fraction
.12
.1E+00
.8E-01
.6E-01
.4E-01
.2E-01
0
0
1000
2000
3000
4000
5000



AP1000 Transient Steam Release Investigation
Minimum Initial Seconday Mass (MSSG0)
Base Case
Reduced TSMAX
Reduced SGMSCR
Reduced TSMAX & SGMSCR
Power Transient (with reduced TSMAX)
Power Transient (with reduced SGMSCR)
Tavg Flashing Fraction
.12
.1E+00
.8E-01
.6E-01
.4E-01
.2E-01
0
0
1000
2000
3000
4000
5000

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.003 (Response Revision 1)

Question:

Please provide the following information with regard to the radiological consequences analysis of the design-basis Rod Ejection Accident (REA) as discussed in Chapter 15.4.8.3 and Table 15.4-4 of the AP1000 DCD:

A. A fraction of the fuel rods are assumed to melt in the radiological analysis of the REA. Regulatory Position 3 of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," states that, for design-basis accident events that do not assume melting of the entire core, radial peaking factors should be applied in determining the inventory of the damaged rods. This does not appear to have been done. Please either update your analysis to include the maximum radial peaking factor in the determination of the source term if it was not included, or provide a basis for why you did not do so.

B. What is the basis for the assumed leak flashing fraction of 4.0% in the radiological consequences analysis of the REA?

C. What assumptions were made in the determination of the steam mass release from the secondary system assumed in the radiological consequences analysis of the REA? What is the basis for the assumed release duration of 1800 seconds?

D. What is the basis for the alkali metal partition coefficient of 0.001 used in the REA radiological consequences analysis? What assumptions were made in the determination of the value?

Westinghouse Response:

A. The analysis included the maximum radial peaking factor of 1.65 in the calculation of activity released from failed/melted fuel. DCD, Chapter 15.4, Table 15.4-4 will be modified, as indicated below, to reflect this assumption.

B. As discussed below (Item C) the SBLOCA transient was used to provide transient data for the rod ejection radiological consequences analysis. The flashing fraction was calculated using the transient vessel average temperature from the SBLOCA analysis. The fraction of 0.04 (4% flashing) was chosen to bound the transient results. The analysis conservatively maintained this fraction for the initial 1800 seconds of the transient.

C. The design basis rod ejection transient results from a mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. This failure results in a loss of coolant accident with a possible reactivity insertion event. The steam generator steam releases to the environment, the time for the primary pressure to fall below the secondary pressure and the leak flashing fraction were chosen to bound those calculated for the 2 inch small break loss of coolant accident (SBLOCA). The 2 inch break is smaller than the flow area that results from a control rod mechanism pressure housing failure. The smaller break conservatively extends the steam releases and has a slower primary depressurization. This delays the time when the primary pressure drops below the secondary pressure and extends the time when the steam generators are steaming to remove decay heat.

Figure 15.6.5.4B-17 of the DCD shows the primary pressure transient for the SBLOCA. The steam generator pressure is maintained at the safety valve setpoint until the reverse heat transfer to the primary system starts when the primary pressure falls below the secondary pressure. From Figure 15.6.5.4B-17 the primary pressure is below the secondary pressure well before the 1800 seconds assumed in the radiological consequences analysis.

Heat transfer to the steam generators, and consequently steam releases from the steam generators, also stops well before 1800 seconds. The average steam flow rate until steam releases stop was calculated using the SBLOCA analysis results and conservatively increased to 60 lbm/sec for use in the radiological consequences analysis. The analysis conservatively maintained this rate for the initial 1800 seconds of the transient, resulting in a total steam release of 1.08E5 lbm.

D. The retention of particulate radionuclides such as alkali metals in the steam generators is limited by the moisture carryover from the steam generators consistent with the guidance of RG 1.183. The design full power moisture carryover fraction for the AP1000 is 0.001, and the moisture carryover would drop following reactor trip. The radiological consequences analysis conservatively maintained the full power value for the duration of the analysis.

Design Control Document (DCD) Revision:

Table 15.4-4 (Sheet 1 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g of dose equivalent I-131 (see Appendix 15A)[(a)]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g dose equivalent **Xe 133**
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of reactor coolant concentrations at maximum equilibrium conditions
Radial peaking factor (for determination of activity in failed/melted fuel)	**1.65**
Fuel cladding failure	
– Fraction of fuel rods assumed to fail	0.1
– Fission product gap fractions	
Iodines and noble gases	0.1
Alkali metals	0.12
Core melting	
– Fraction of core melting	0.0025
– Fraction of activity released	
Iodines and alkali metals	0.5
Noble gases	1.0
Iodine chemical form (%)	
– Elemental	4.85
– Organic	0.15
– Particulate	95.0
Core activity	See Table 15A-3 in Appendix 15A
Nuclide data	See Table 15A-4 in Appendix 15A
Reactor coolant mass (lb)	3.7 E+05

PRA Revision:

None

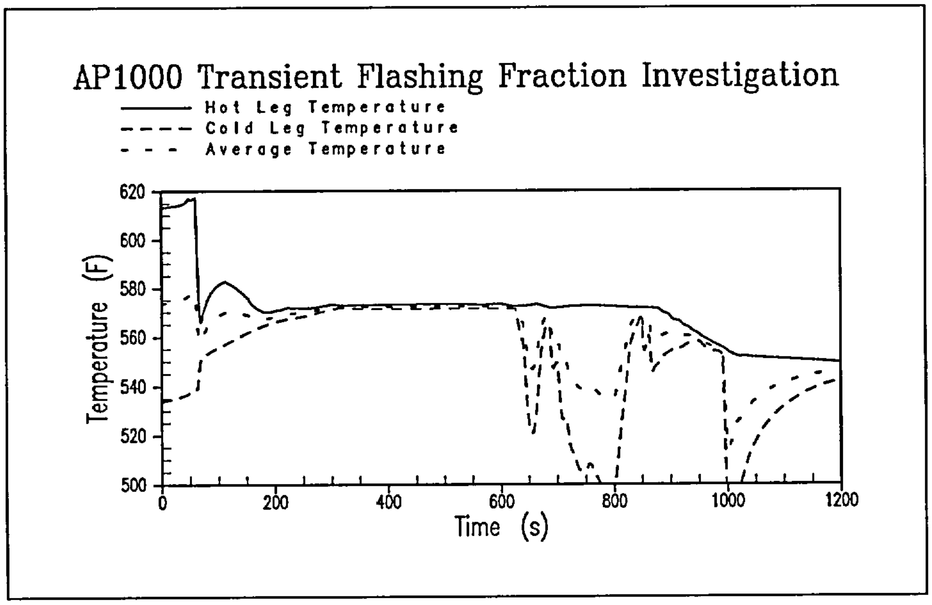
NRC Additional Comment:

B. What is the basis for the assumed leak flashing fraction of 4.0% in the radiological consequences analysis of the REA?

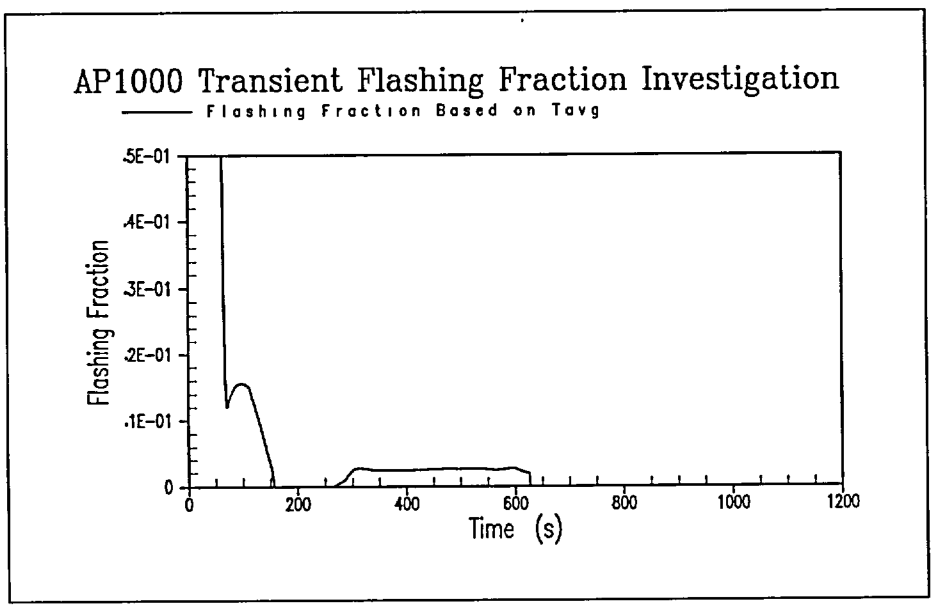
The Westinghouse response states that the flashing fraction was calculated using the transient vessel average temperature from the small break loss of coolant analysis (SBLOCA) analysis. What value was used? Is this noted somewhere in the DCD?

Westinghouse Additional Response:

Figure 15.6.5.4B-17 of the DCD shows the primary pressure transient for the SBLOCA. The corresponding reactor coolant system temperatures and secondary pressure were obtained for use in calculating the flashing fraction. The figures below provide the reactor coolant temperatures and the corresponding calculated flashing fraction based on the average reactor coolant temperature. The bounding flashing fraction of 0.04 was initially selected based on preliminary analysis results and retained even though the DCD analysis indicates that a lower flashing fraction is justified. This background information is not included in the DCD.



AP1000 Transient Flashing Fraction Investigation
Hot Leg Temperature
Cold Leg Temperature
Average Temperature
Temperature (F)
620
600
580
560
540
520
500
0
200
400
600
800
1000
1200
Time (s)



AP1000 Transient Flashing Fraction Investigation
Flashing Fraction Based on Tavg
Flashing Fraction
.5E-01
.4E-01
.3E-01
.2E-01
.1E-01
0
200
400
600
800
1000
1200
Time (s)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.007 (Response Revision 1)

Question:

All Chapter 15 design-basis accident radiological analyses include a discussion of additional radiological consequences of spent fuel pool boiling that may occur coincident with the accident. What assumptions and inputs were used to calculate the radiological consequences as a result of spent fuel pool boiling?

Westinghouse Response:

The following assumptions and inputs specific to modeling the activity release from spent fuel boiling were used:

Parameter	Value	Notes
Initial activity in spent fuel pool I-131 Other nuclides	 3.18 Ci None modeled	It is conservatively assumed that all activity is I-131. The activity is based on the concentration that will result in a radiation field of 2.5 mrem/hr at the pool surface.
Fuel stored in the spent fuel pool	See note	It is assumed that the spent fuel from ten years of operation is in the spent fuel racks, including a region (68 fuel assemblies) from a recent refueling. Based on a nominal 18 month fuel cycle, there are regions that have the following decay intervals: 399 hours, 1.5 years, 3 years, 4.5 years, 6 years, 7.5 years, and 9 years.
Amount of I-131entering the pool over a 30-day period due to diffusion from fuel rods containing cladding defects	1.94 Ci	Although the release of activity to the water pool takes place over the 30 day interval, the activity is conservatively assumed to all be present in the pool at the onset of pool boiling.
Initial pool water temperature	120 °F	
Time to initiate pool boiling	8.8 hr	

Parameter	Value	Notes
Steaming rate 8.8 hr 24 hr 48 hr 72 hr 88 hr 120 hr ≥168	 16,200 lb/hr 16,000 15,700 15,420 15,250 14,930 14,500	Continued reduction in decay heat generation after 168 hours (7 days) was not modeled. The steaming rates conservatively assume that all heat loss is through steaming. Credit was not taken for termination of steaming after 7 days although support from offsite could be credited at that point.
Partition coefficient for iodine	0.01	
Spent fuel pool water mass 8.8 hr 24 hr 48 hr 72 hr ≥88 hr	 1.19E6 lb 1.05E6 lb 8.50E5 lb 6.16E5 lb 4.40E5 lb	Although makeup water would be available earlier, no credit is taken until 88 hours when the water level has dropped to the top of the stored fuel assemblies. The water level in the pool is assumed to be maintained at that level with no credit for the makeup water increasing the level.
Atmospheric dispersion factors at LPZ boundary 8 – 24 hr 24 – 96 hr >96 hr	 1.0E-4 sec/m^3 5.4E-5 sec/m^3 2.2E-5 sec/m^3	There are no releases modeled prior to 8.8 hours when boiling starts.
Atmospheric dispersion factors at CR intake 8 – 24 hr 24 – 72 hr 72 – 96 hr >96 hr	 3.0E-4 sec/m^3 3.0E-4 sec/m^3 1.0E-3 sec/m^3 9.0E-4 sec/m^3	There are no releases modeled prior to 8.8 hours when boiling starts. The intake point is initially the entrance to the control room. After 72 hours the intake point is the normal air intake.
Nuclide data	See DCD Table 15A-4	
Offsite breathing rate 8 – 24 hr >24 hr	 1.8E-4 m^3/sec 2.3E-4 m^3/sec	

Additional assumptions associated with modeling the control room are as described for the LOCA dose analysis (DCD Table 15.6.5-2).

NRC Additional Comments:

A. The Westinghouse response states that the initial activity in the spent fuel pool is 3.18 Ci of I-131, and that it is based on a concentration that will result in a radiation field of 2.5 mrem/hr at the pool surface. Why was 2.5 mrem/hr chosen? How did you determine the activity of I-131 that results in 2.5 mrem/hr at the pool surface? Please provide details.

B. The Westinghouse response states that the amount of I-131 entering the pool over a 30-day period due to diffusion from fuel rods containing cladding defects is 1.94 Ci. How did you determine this value?

Westinghouse Additional Response:

A) The basis for a radiation field of 2.5 mrem/hr at the pool surface is the long-standing Westinghouse recommendation to allow continuous 40 hours per week operator occupancy.

The I-131 concentration that would result in a pool surface radiation field of 2.5 mrem/hr was determined to be 5.9E-3 μCi/gm based on standard point-kernel calculations. The concentration was then converted to inventory based on the mass of water assumed to be in the pool.

B) The determination of the amount of I-131 entering the pool over a 30-day period due to diffusion from the fuel rods containing cladding defects was conservatively calculated using the following assumptions:

- One third of a reactor core is present in the spent fuel pool with only 100 hours of decay (this neglects the additional decay that occurred during the refueling process).
- The fuel assemblies are assumed to have been operating at average core power.
- The design basis fuel defect level of 0.25% is assumed to apply to the stored fuel and no credit is taken for the closure of small cladding defects that would be expected to occur after the fuel has cooled down from normal operating conditions.
- An I-131escape rate coefficient of 1.3E-13 sec^{-1} is used for the fuel while stored in the pool. This is a factor of 1.0E5 lower than the coefficient of 1.3E-8 sec^{-1} assumed during full power operation. [The value of 1.3E-8 sec^{-1} was selected in Reference 1 to bound the value of 2.0E-9 sec^{-1} that was determined in the Reference 1 test evaluation. The value also bounds the value of 1.0E-8 sec^{-1} identified in Reference 2.]
- No decay of the I-131source term beyond 100 hours was credited.

Using these assumptions, there is a contained source term of 2.3E7 Ci of I-131 in the fuel and a total release of 0.0194 Ci to the pool over 30 days. For additional conservatism, this amount was increased by a factor of 100.

Reference: 1. W. D. Fletcher and L. F. Picone, "Fission Products from Fuel Defect Test at Saxton," WCAP-3269-63, prepared for the U. S. Atomic Energy Commission by Westinghouse Electric Corp., Atomic Power Division (April 1966)
2. P. Cohen and T. J. Iltis, Chapter 7 of *The Shippingport Pressurized Water Reactor,* US Atomic Energy Commission, Naval Reactors Branch; Westinghouse Electric Corporation, Bettis Plant; and Duquesne Light Company, pp 181-201, Addison-Wesley Publishing Company, Inc., Reading, MA (1958).

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 720.058 (Response Revision 1)

Question:

Westinghouse claims that the concrete penetration on the reactor coolant drain tank (RCDT) (sump) side of the cavity is minimal following a hinged failure mode of the reactor vessel (RV), compared to the penetration on the RV side of the cavity. However, this is predicated on the core debris separating, with the oxide component (about 85 - 90 percent oxide) remaining on the reactor vessel side of the cavity, and a metallic component (about 75 to 85 percent metal) reaching the RCDT side of the cavity. This debris separation behavior is used by Westinghouse as the basis for concluding that core debris accumulation in the cavity sump would not be controlling for basemat melt-through. It is unclear whether this separation will actually occur given the large uncertainties in the configuration of molten core debris prior to vessel breach (i.e., mixed versus stratified), and the turbulence and mixing that would occur as the debris enters and spreads within the reactor cavity. Please confirm the robustness of your conclusion and the adequacy of the sump curb design by providing an assessment of the impact on basemat melt-through times and containment pressure (for both limestone and basaltic concretes) assuming that this oxide/metallic separation does not occur following a hinged failure of the reactor vessel, i.e., either a homogeneous melt or an oxide melt reaches the RCDT side of the reactor cavity and enters the sump.

Westinghouse Response:

The MELTSPREAD analyses that were performed for the two vessel failure scenarios determined the partitioning of the core debris between the reactor cavity and the reactor coolant drain tank room. The bounds of debris quenching and spreading during relocation are included in the investigation.

The masses of the debris released to the cavity from the hinged vessel failure are presented in Table B-1 of the AP1000 PRA report. The masses are summarized here in Table 1 and presented as the components uranium dioxide, zirconium, zirconium dioxide, and stainless steel.

Table 1 – Debris Volume in the Cavity

Debris Component	Mass (kg)	Volume (m^3)
UO_2	96,500	11.0
Zr	14,755	2.4
ZrO_2	10,726	1.8
SS	51,000	7.3
Total Volume		22.5

The cavity floor surface area is 48.3 m^2. If the debris is uniformly spread over the surface of the cavity floor, the debris depth is 47 cm or 18.3 inches. The height of the curb currently is 18 inches. The height of the curb will be increased to 24 inches (see PRA revision below) to prevent debris from entering the sump even for this non-mechanistic case.

Design Control Document (DCD) Revision:

None

PRA Revision:

B.4 Core Concrete Interactions

If the reactor vessel fails when the RCS is at a low pressure, the molten core debris will pour from the reactor vessel onto the reactor cavity floor. If a steam explosion does not occur, the pour will spread over the cavity floor and begin to transfer heat to the concrete floor of the reactor cavity. Due to the predicted mode of reactor vessel failure and the shape of the AP600 reactor cavity, analyses of the possible spreading of the core debris over the cavity floor were conducted using the MELTSPREAD code (Reference B-4). The AP1000 cavity geometry is the same as AP600. In addition, the AP1000 initial debris location from the vessel to the cavity is similar to AP600 in terms of mass flowrate and superheat, and therefore, the MELTSPREAD analyses performed for AP600 can be extended to AP1000. The results of the MELTSPREAD analyses were used as input to the MAAP4 code for analysis of core concrete interactions for AP1000.

The AP1000 reactor cavity is at containment elevation 71′ 6″ and consists of two interconnected volumes. The volume, which includes the reactor vessel, is octagonal in shape. The other volume is rectangular in shape and houses the reactor coolant drain tank (RCDT) and also contains the reactor cavity sump. The two volumes are connected by a 5-foot wide tunnel whose floor is also at elevation 71′ 6″ and a 3-foot wide ventilation duct whose bottom is 4 inches above the cavity floor. The cavity sump is situated between the tunnel and the ventilation duct at the side of the reactor coolant drain tank room closest to the reactor vessel. There is a 3-foot thick wall that separates the reactor cavity drain tank region from the reactor vessel region of the cavity. The floor of the cavity sump is at elevation 69′ 6″ and is completely encompassed by a curb whose top is at elevation 73′ 6″ (24-inch high curb). The tunnel between the reactor vessel and reactor coolant drain tank portions of the cavity is protected by a door and shielding material to minimize radiation exposure to persons working in the reactor coolant drain tank area of the cavity. The door and shielding are not important to the analyses of core debris spreading in the reactor cavity due to the dynamic forces of the fuel coolant interactions that will occur at reactor vessel failure. Since the door and shielding are not designed to withstand "blast loading," they are expected to be destroyed prior to the arrival of core debris at their pre-vessel failure location. As added assurance that the door and shielding will not remain in their pre-vessel failure location, the high temperature of the core debris will quickly ablate and/or physically move any door and/or shielding components that might remain in place after the fuel coolant interaction loading. A schematic layout of the cavity region is provided in Figure B-3.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

NRC Additional Comments:

The RAI requested that W provide an assessment of the impact on basemat melt-through times and containment pressure (for both limestone and basaltic concretes) assuming that oxide/metallic separation does not occur, in order to confirm their conclusion regarding basemat failure and the adequacy of the sump curb design. In their response, W indicated that the sump curb height will be increased, but they did not provide the requested assessment.

Revise response to point to the EQ survivability CCI case. Make the case.

Westinghouse Additional Response:

A MAAP4 analysis of core-concrete interaction in the AP1000 cavity considering a uniform debris bed is provided in Appendix D of PRA as part of the MAAP4 runs supporting equipment survivability environments. The following results in Table 1 and in Figures 1 and 2 are taken from those analyses.

Table 1 - Core Concrete Interaction Results Uniform Debris Spreading		
Parameter	Basaltic Concrete	Limestone-Sand Concrete
Time of embedded shell melt-through	6.3 hours	7.1 hours
Time of basemat melt through	53 hours (2.2 days)	77 hours (3.2 days)
Pressure at 24 hours	14.1 psig	25.3 psig
Pressure at basemat melt through	26 psig	50.5 psig

The results, while showing slightly faster times to basemat melt-through, do not fail the containment within 24 hours, and thus do not impact the conclusions to the analysis presented in Appendix B of the PRA report.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

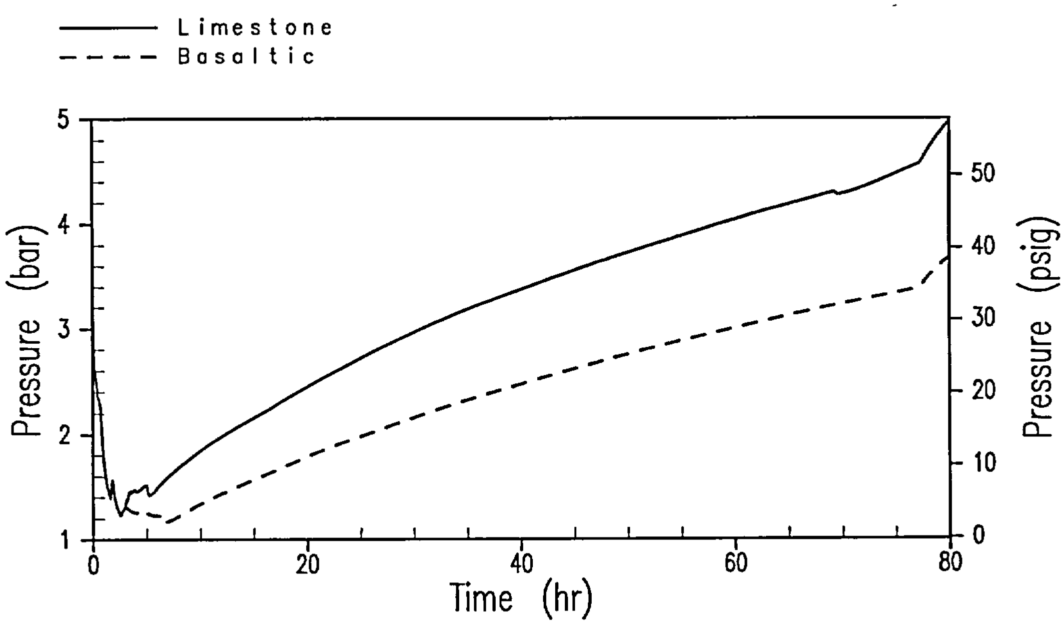


Figure 1
AP1000 Concrete Interaction with Uniform Debris Spreading
Containment Pressure

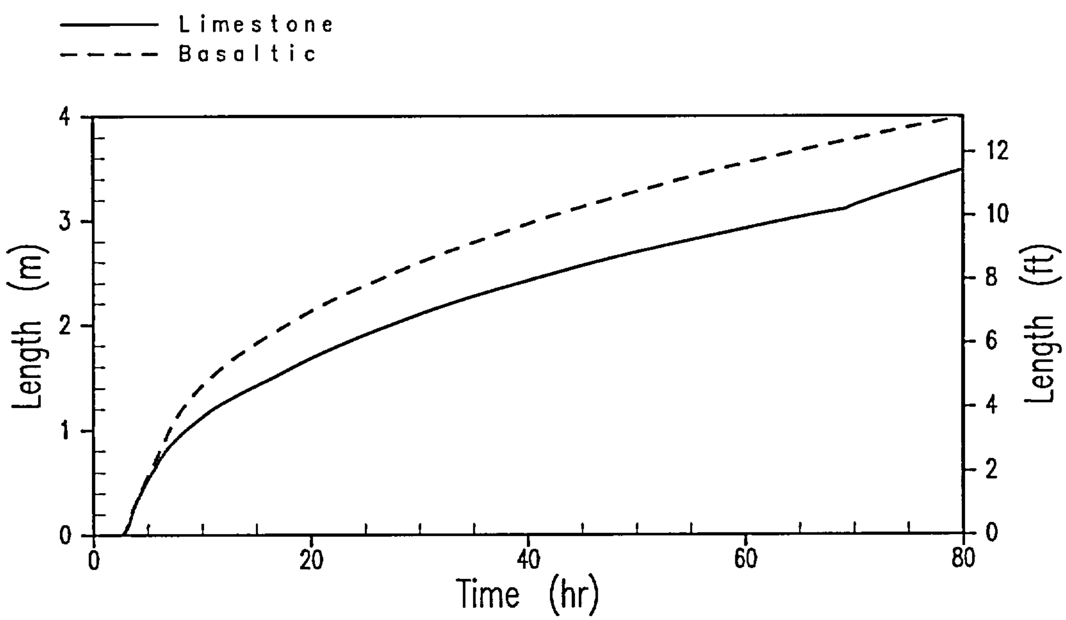


Figure 2
AP1000 Concrete Interaction with Uniform Debris Spreading
Downward Basemat Penetration

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.082 (Response Revision 1)

Question:

In order to obtain a clearer picture of severe accident progression in the AP1000, the staff plans to perform confirmatory MELCOR analyses for several sequences that are either dominant overall in terms of CDF or dominant within some set of risk-significant sequences. The AP1000 PRA, page 59-7, describes in detail the 5 sequences with highest CDF. Please provide a similarly detailed description of the following additional sequences:

A. Sequence 13 (SGTR initiator, failure of CMT or RCP trip, success of PRHR, failure of full and partial ADS). This is the highest-frequency SGTR-initiated core damage sequence reported.

B. Sequence 20 (Transient, failure of MFW/SFW/PRHR [main feedwater system/startup feedwater system/passive residual heat removal system], success of core makeup tank [CMT] and reactor coolant pump [RCP] trip, failure of full and partial automatic despressurizatioin system [ADS]). This is the highest-frequency non-bypass sequence expected to be at high RCS pressure at the time that core damage begins.

Westinghouse Response:

The sequence descriptions for the dominant AP1000 PRA CDF for at-power events sequences # 13 and # 20 are given below.

13. SEQUENCE 6ESGT-41

A SG tube rupture event occurs. Passive RHR system is successful but the CMT injection to make up RCS inventory loss fails. Full and partial ADS also fail. Thus, the RCS makeup by IRWST or normal NRHR can not be provided. Early core damage is postulated, leading to a containment bypass end state 6. The sequence frequency is 3.55E-09/yr, contributing 1.47% to the plant core damage frequency.

20. SEQUENCE 1ATRA-17

A transient with MFW available initiating event occurs. During the event, main feedwater and startup feedwater fail. Passive RHR also fails. CMT injection is successful but both full and partial ADS fail. Early core damage with high RCS pressure is postulated, leading to the end state 1A. The sequence frequency is 1.41E-09/yr, contributing 0.59% to the plant core damage frequency.

The dominant CDF cutsets for the two sequences are given in Tables 720.082-1 and 2.

Table 720.082-1 AP1000 PRA 6ESGT-41 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
1	7.58E-10	21.32	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			COGNITIVE OPERATOR ERROR	1.84E-03	CIB-MAN00
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO RECOG. THE NEED FOR RCS DEPRESS. DURING SLOCA	5.00E-01	LPM-MAN01C
2	7.58E-10	21.32	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			COGNITIVE OPERATOR ERROR	1.84E-03	CIB-MAN00
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			COND. PROB. OF ADN-MAN01(OPER. FAILS TO ACT. ADS)	5.00E-01	ADN-MAN01C
3	6.39E-10	17.97	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			FAILURE TO ALIGN CVCS IN AUX. SPRAY MODE	3.10E-03	CVN-MAN00
			OPERATOR FAILS TO FULFIL MANUAL ACTUATION OF ADS	5.00E-01	ADF-MAN01
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			COND. PROB. OF ADN-MAN01(OPER. FAILS TO ACT. ADS)	5.00E-01	ADN-MAN01C
4	5.52E-10	15.53	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			OPERATOR ERROR TO CLOSE VALVES ON RUPTURED SG	1.34E-03	CIB-MAN01
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO RECOG. THE NEED FOR RCS DEPRESS. DURING SLOCA	5.00E-01	LPM-MAN01C
5	1.12E-10	3.15	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR

Table 720.082-1 AP1000 PRA 6ESGT-41 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
			COGNITIVE OPERATOR ERROR	1.84E-03	CIB-MAN00
			COMMON CAUSE FAILURE OF 4 AOVS TO OPEN	6.20E-05	CCX-AV-LA
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			COND. PROB. OF ADN-MAN01(OPER. FAILS TO ACT. ADS)	5.00E-01	ADN-MAN01C
6	9.46E-11	2.66	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			FAILURE TO ALIGN CVCS IN AUX. SPRAY MODE	3.10E-03	CVN-MAN00
			OPERATOR FAILS TO FULFIL MANUAL ACTUATION OF ADS	5.00E-01	ADF-MAN01
			COMMON CAUSE FAILURE OF 4 AOVS TO OPEN	6.20E-05	CCX-AV-LA
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			COND. PROB. OF ADN-MAN01(OPER. FAILS TO ACT. ADS)	5.00E-01	ADN-MAN01C
7	9.24E-11	2.6	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			COGNITIVE OPERATOR ERROR	1.84E-03	CIB-MAN00
			COMMON CAUSE FAILURE OF 4 CHECK VALVES TO OPEN	5.10E-05	CMX-CV-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			COND. PROB. OF ADN-MAN01(OPER. FAILS TO ACT. ADS)	5.00E-01	ADN-MAN01C
8	7.73E-11	2.17	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			FAILURE TO ALIGN CVCS IN AUX. SPRAY MODE	3.10E-03	CVN-MAN00
			OPERATOR FAILS TO FULFIL MANUAL ACTUATION OF ADS	5.00E-01	ADF-MAN01
			COMMON CAUSE FAILURE OF 4 CHECK VALVES TO OPEN	5.10E-05	CMX-CV-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			COND. PROB. OF ADN-MAN01(OPER. FAILS TO ACT. ADS)	5.00E-01	ADN-MAN01C
9	3.59E-11	1.01	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR

Table 720.082-1 AP1000 PRA 6ESGT-41 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
			MECHANICAL FAILURE OF AOV V084 AND CV V085 TO OPEN	2.88E-02	CVMOD05
			OPERATOR FAILS TO FULFIL MANUAL ACTUATION OF ADS	5.00E-01	ADF-MAN01
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
10	3.38E-11	0.95	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			MECHANICAL FAILURE OF AOV V081 FAILS TO CLOSE	2.71E-02	CVMOD07
			OPERATOR FAILS TO FULFIL MANUAL ACTUATION OF ADS	5.00E-01	ADF-MAN01
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
11	1.59E-11	0.45	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			MECHANICAL FAILURE OF AOV V084 AND CV V085 TO OPEN	2.88E-02	CVMOD05
			OPERATOR FAILS TO FULFIL MANUAL ACTUATION OF ADS	5.00E-01	ADF-MAN01
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO RECOG. THE NEED FOR RCS DEPRESS. DURING SLOCA	1.34E-03	LPM-MAN01
12	1.50E-11	0.42	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			COGNITIVE OPERATOR ERROR	1.84E-03	CIB-MAN00
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			FAILURE OF MANUAL DAS REACTOR TRIP HARDWARE	1.00E-02	MDAS
			COND. PROB. OF ADN-MAN01(OPER. FAILS TO ACT. ADS)	5.00E-01	ADN-MAN01C

Table 720.082-1 AP1000 PRA 6ESGT-41 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
13	1.50E-11	0.42	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			COGNITIVE OPERATOR ERROR	1.84E-03	CIB-MAN00
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			FAILURE OF MANUAL DAS REACTOR TRIP HARDWARE	1.00E-02	MDAS
			OPER. FAILS TO RECOG. THE NEED FOR RCS DEPRESS. DURING SLOCA	5.00E-01	LPM-MAN01C
14	1.50E-11	0.42	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			MECHANICAL FAILURE OF AOV V081 FAILS TO CLOSE	2.71E-02	CVMOD07
			OPERATOR FAILS TO FULFIL MANUAL ACTUATION OF ADS	5.00E-01	ADF-MAN01
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO RECOG. THE NEED FOR RCS DEPRESS. DURING SLOCA	1.34E-03	LPM-MAN01
15	1.30E-11	0.37	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			FAILURE OF AIR COMPRESSOR TRANSMITTER	5.23E-03	CANTP011RI
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
16	1.27E-11	0.36	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			MAIN GEN. BKR ES 01 FAILS TO OPEN [#	5.08E-03	EC0MOD01
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
17	1.26E-11	0.35	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR

Table 720.082-1 AP1000 PRA 6ESGT-41 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
			FAILURE TO ALIGN CVCS IN AUX. SPRAY MODE	3.10E-03	CVN-MAN00
			OPERATOR FAILS TO FULFIL MANUAL ACTUATION OF ADS	5.00E-01	ADF-MAN01
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			FAILURE OF MANUAL DAS REACTOR TRIP HARDWARE	1.00E-02	MDAS
			COND. PROB. OF ADN-MAN01(OPER. FAILS TO ACT. ADS)	5.00E-01	ADN-MAN01C
18	1.09E-11	0.31	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			OPERATOR ERROR TO CLOSE VALVES ON RUPTURED SG	1.34E-03	CIB-MAN01
			COMMON CAUSE FAILURE TO OPEN OF 4.16 KVAC CIRCUIT BREAK	4.20E-04	RPX-CB-GO
			FAILURE OF MANUAL DAS REACTOR TRIP HARDWARE	1.00E-02	MDAS
			OPER. FAILS TO RECOG. THE NEED FOR RCS DEPRESS. DURING SLOCA	5.00E-01	LPM-MAN01C
19	7.46E-12	0.21	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
			PUMP B FAILS TO TRIP - BREAKER FAILS TO OPEN	4.20E-03	RC1CB063GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
20	7.46E-12	0.21	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
			PUMP B FAILS TO TRIP - BREAKER FAILS TO OPEN	4.20E-03	RC1CB061GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
21	7.46E-12	0.21	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM

Table 720.082-1 AP1000 PRA 6ESGT-41 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
			PUMP A FAILS TO TRIP - BREAKER FAILS TO OPEN	4.20E-03	RC1CB053GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
22	7.46E-12	0.21	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
			PUMP A FAILS TO TRIP - BREAKER FAILS TO OPEN	4.20E-03	RC1CB051GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
23	7.46E-12	0.21	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3 88E-03	IEV-SGTR
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
			PUMP B FAILS TO TRIP - BREAKER FAILS TO OPEN	4.20E-03	RC1CB063GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
24	7.46E-12	0.21	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
			PUMP B FAILS TO TRIP - BREAKER FAILS TO OPEN	4.20E-03	RC1CB061GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
25	7.46E-12	0.21	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	3.88E-03	IEV-SGTR
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
			PUMP A FAILS TO TRIP - BREAKER FAILS TO OPEN	4.20E-03	RC1CB053GO
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACTUATION)	5.06E-01	REC-MANDASC

Table 720.082-1 AP1000 PRA 6ESGT-41 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01

Table 720.083-2 AP1000 PRA 1ATRA-17 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
1	2.28E-10	16.12	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			FAILURE OF AIR COMPRESSOR TRANSMITTER	5.23E-03	CANTP011RI
			FAILURE OF IRWST GUTTER DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA1
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
2	2.28E-10	16.12	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			FAILURE OF AIR COMPRESSOR TRANSMITTER	5.23E-03	CANTP011RI
			FAILURE OF PRHR DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
3	1.18E-10	8.35	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC1BS001TM
			FAILURE OF IRWST GUTTER DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA1
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
4	1.18E-10	8.35	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC1BS001TM
			FAILURE OF PRHR DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
5	1.18E-10	8.35	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS012TM
			FAILURE OF IRWST GUTTER DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA1
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO

Table 720.083-2 AP1000 PRA 1ATRA-17 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
6	1.18E-10	8.35	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS012TM
			FAILURE OF PRHR DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
7	6.61E-11	4.67	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF NON-SAFETY TRANSMITTERS INTERFACING SYSTEM PRESSU	4.78E-04	CCX-TRNSM
			EDS3 EA 1 DISTR. PNL FAILURE OR T&M	3.05E-04	ED3MOD07
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
8	2.54E-11	1.8	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF NON-SAFETY TRANSMITTERS INTERFACING SYSTEM PRESSU	4.78E-04	CCX-TRNSM
			CCF OF TEMPERATURE TRANSMITTERS (CCX-TT-UF)	1.17E-04	CCX-TT-UF
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
9	2.54E-11	1.8	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF OF TEMPERATURE TRANSMITTERS (CCX-TT-UF)	1.17E-04	CCX-TT-UF
			CCF OF SAFETY PT LT CONTINUOSLY INTERFACING HIGH PRESSURE	4.78E-04	CCX-XMTR
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
10	2.52E-11	1.78	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF NON-SAFETY TRANSMITTERS INTERFACING SYSTEM PRESSU	4.78E-04	CCX-TRNSM
			UNAVAILABILITY GOAL FOR DAS	1.00E-02	DAS
			FAILURE OF MANUAL DAS ACT.	1.16E-02	REC-MANDAS
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO

Table 720.083-2 AP1000 PRA 1ATRA-17 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
11	2.17E-11	1.53	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF NON-SAFETY TRANSMITTERS INTERFACING SYSTEM PRESSU	4.78E-04	CCX-TRNSM
			UNAVAILABILITY GOAL FOR DAS	1.00E-02	DAS
			FAILURE OF MANUAL DAS REACTOR TRIP HARDWARE	1.00E-02	MDAS
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
12	2.08E-11	1.47	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF NON-SAFETY TRANSMITTERS INTERFACING SYSTEM PRESSU	4.78E-04	CCX-TRNSM
			FAILURE OF IRWST GUTTER DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA1
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
13	2.08E-11	1.47	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF NON-SAFETY TRANSMITTERS INTERFACING SYSTEM PRESSU	4.78E-04	CCX-TRNSM
			FAILURE OF PRHR DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
14	1.10E-11	0.78	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF OF SUB-SYSTEMS IN SIGNAL SELECTOR CABINET	2.53E-04	CCX-PLSMOD6
			FAILURE OF IRWST GUTTER DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA1
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
15	1.10E-11	0.78	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF OF SUB-SYSTEMS IN SIGNAL SELECTOR CABINET	2.53E-04	CCX-PLSMOD6
			FAILURE OF PRHR DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO

Table 720.083-2 AP1000 PRA 1ATRA-17 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
16	1.10E-11	0.78	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			OPERATING BLOWER FAN HARDWARE FAILURE	2.52E-04	SWAMOD09T
			FAILURE OF IRWST GUTTER DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA1
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
17	1.10E-11	0.78	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			OPERATING BLOWER FAN HARDWARE FAILURE	2.52E-04	SWAMOD09T
			FAILURE OF PRHR DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
18	1.02E-11	0.72	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			MAIN GEN. BKR ES 01 FAILS TO OPEN [#	5.08E-03	EC0MOD01
			STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO1DG001TM
			FAILURE OF IRWST GUTTER DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA1
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
19	1.02E-11	0.72	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			MAIN GEN. BKR ES 01 FAILS TO OPEN [#	5.08E-03	EC0MOD01
			STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO1DG001TM
			FAILURE OF PRHR DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
20	7.63E-12	0.54	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF NON-SAFETY TRANSMITTERS INTERFACING SYSTEM PRESSU	4.78E-04	CCX-TRNSM
			UNAVAILABILITY GOAL FOR DAS	1.00E-02	DAS
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS AC	5.06E-01	REC-MANDASC

Table 720.083-2 AP1000 PRA 1ATRA-17 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
			3/4 STAGE 2 & 3 LINES FAIL DUE TO CCF OF MOVs TO OPEN	7.48E-04	ADX-MV-GO
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
21	6.14E-12	0.43	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PL9MOD1
			FAILURE OF IRWST GUTTER DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA1
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
22	6.14E-12	0.43	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PL9MOD1
			FAILURE OF PRHR DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3 24E-04	ADX-MV3-GO
23	5.69E-12	0.4	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			FAILURE OF AIR COMPRESSOR TRANSMITTER	5.23E-03	CANTP011RI
			PLUG/LEAK OF PRHR HEAT EXCHANGER	2.40E-06	PCNHR001ML
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
24	5.69E-12	0.4	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			FAILURE OF AIR COMPRESSOR TRANSMITTER	5.23E-03	CANTP011RI
			FAILURE OF THE PRHR DUE TO IRWS TANK FAILURE	2.40E-06	IWNTK001AF
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO
25	5.23E-12	0.37	TRANSIENT WITH MFW INITIATING EVENT OCCURS	1.40E+00	IEV-TRANS
			COMMON CAUSE FAILURE OF COMPRESSORS TO RUN	1.20E-04	CAX-CM-ER
			FAILURE OF IRWST GUTTER DUE TO COMMON CAUSE OF AOVs	9.60E-05	PXX-AV-LA1

Table 720.083-2 AP1000 PRA 1ATRA-17 SEQUENCE DOMINANT CDF CUTSETS

	CUTSET FREQ.PROB	% of SEQ. CDF	BASIC EVENT NAME	EVENT PROB.	BASIC EVENT IDENTIFIER
			CCF OF 4 COMBINATIONS OF 3 STAGES #2 AND #3 MOVS	3.24E-04	ADX-MV3-GO

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

Please clarify the following aspects of sequence #20 (1ATRA-17) of the AP1000 PRA. If the cutsets comprising this sequence include substantial relative contributions with different characteristics regarding these points, also specify approximate frequency contributions.

a. Although IVR by means of lower head cooling cannot be credited for this high-pressure sequence, what is the status of cavity flooding from the IRWST?

b. Are the gutter drain valves assumed to close successfully? (i.e., is condensate from the containment directed to the containment sump or to the IRWST in this scenario?)

Westinghouse to revise the RAI response.

Westinghouse Additional Response:

a. The equipment for cavity flooding is available for this sequence. However, operator action (which is already proceduralized) must be performed to make the cavity flooding happen. For this sequence the cavity flooding is modeled in the containment event tree by two different means:

1. (DP node of the containment event tree) If the later operator action to depressurize the RCS is successful, the cavity will be flooded as a consequence of the depressurization path and IRWST injection. A calculation for the success of this event tree node for this specific sequence shows that the depressurization would be successful (the cavity will be flooded) 86.4% of the time, and depressurization will fail in 13.6% of the time.

2. (IR node of the containment event tree) Even if the DP node fails, the operators are instructed at a later time to flood the cavity. The equipment is available. The failure probability of the cavity flooding (IWF) is 7.9E-03, which is very low. However, it is dependent on an operator action. Failure of DP is dominated by failure of an earlier operator action. Since these two operator actions are in the same phase of the accident progression (their cues come from the core exit thermocouple temperatures, 700 and 1200 degrees F), the AP1000 containment model does not credit cavity flooding if DP node fails.

   If even HEP = 0.1 credit is given for the operator action of implementing the cavity flooding after the failure of late RCS depressurization (DP node), the overall success fraction of the cavity flooding will become 98.6% (failure fraction would be 1.4%) for this sequence.

As a conclusion, the cavity flooding is estimated to be implemented in 86.4% of the time for this sequence as a conservative estimate; its success may be as high as 98.6%.

b. An examination of the cutsets of the CDF file for sequence 1ATRA-17, shows that 41% of the CDF is due to the common cause failure of the gutter drain valves to close. This is confirmed by the Fussel-Vesely importance measure of the basic event PXX-AV-LA1. A much smaller contribution exists due to various combinations of the random failures of the two valves (AOV-130A and AOV-130B).

Thus, 59% of the time, the condensate from the containment is directed to the IRWST, and 41% of the time it will go to the sump, in this scenario.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 720.085 (Response Revision 1)

Question:

The AP600 in-vessel steam explosion analysis neglects the possibility of initially small FCIs (with little energetic potential) being a driver for larger melt crucible failures that would increase the melt pour rate. How were these events considered or bounded for the RPV survival in-vessel? Please elaborate.

Westinghouse Response:

The response to this question is based on the following key factors:

a) The reactor vessel lower internals assembly, which includes the core barrel and core support plate, are at the time of interest, still integral and structurally strong. These constitute the outer envelope of the crucible that contains the melt. Only the uppermost area has melted, but we are interested in the lower part. Also, the lower support structure is integral and structurally strong.

b) The downcomer cross sectional area is nearly 4 m^2 and allows relatively free venting up and through the cold legs. This would prevent pressurization during premixing. Also in the event of any significant interaction, with sustained pressures capable to set the lower boundary of the crucible (the crusts), or the crucible as a whole, in motion, this vent area would allow large quantities of lower plenum water to be dispersed, together with venting steam, upwards. Note, in this respect, that only a fraction (-30%) of the core support plate area is open (the flow holes), and also, the inertia mass of the whole lower internals assembly (containing the melt), is at least one order of magnitude greater than any lower plenum water mass coupled in the interaction. This means any pressure developed in between these two masses would tend to expel the water rather than move the core.

c) To fail the lower boundary of the crucible (the crusts), pressure must be applied from below that is high enough and sustained enough to cause motion. This can only be done by forcing water on to this boundary, and this can arise only from a sustained strong interaction in the lower plenum. But an immediate consequence of this is also that another melt-water interaction boundary is formed, at the failing lower boundary of the crucible. This would tend to be self-limiting, as the developing pressure creates a local expansion zone, that again venting downwards, expelling lower plenum water, in a manner that precedes the downward relocation of the melt that would eventually occur. Note that this interaction zone would also contain melt, which would be expelled downwards as well, sustaining the removal of lower plenum water.

d) Throughout all these interactions the structures mentioned under (a) would effectively maintain the retentive property of the crucible, while the core support plate and the internal support structures would effectively prevent a fall-back, gross, contact mechanism. Rather, the fallback would be arrested, and any melt relocation has to occur by gravity, through the holes on the core support plate.

e) By that time hardly any water would have been left in the lower plenum to receive the melt for an explosive interaction. No mechanism that would violate lower head integrity is seen.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

The AP600 in-vessel steam explosion analysis that was cited in support of the AP1000 neglects the possibility of initially small FCIs (with little energetic potential) being a driver for larger melt crucible failures that would increase the melt pour rate. Please elaborate on how these events were considered or bounded for RPV survival in-vessel?

Westinghouse to revise RAI response.

Westinghouse Additional Response:

Based on the discussion of the in-vessel melting and relocation in the AP1000 PRA Chapter 39, revision 1, the bottom crust of in-core debris pool forms approximately 1 meter above the bottom of the active fuel. The lower core support plate, fuel assembly nozzles and the unfueled lower portion of the fuel assemblies are blocked with frozen relocated zirconium and stainless steel, which is cold and structurally sound. The mass of the structure is on the order of 300,000 kg. The mass of the water in the lower plenum is 11000 kg.

An initial steam explosion with a pressure wave large enough to move the lower internals and break the in-core debris pool crucible to create a second, larger debris pour would blow the water out of the lower plenum, as well. Therefore, the pre-mixing of the subsequent debris pour into the lower plenum following the explosion would be water limited. Water remaining in the lower plenum would be highly voided by the heat in the debris initially relocated to the lower plenum. Water draining back onto the lower plenum debris after the initial explosion would not be able to create a premixture as strong as debris pouring into water. Therefore, the analyses performed for the initial debris pour into a lower plenum full of water bounds the steam explosions produced by a debris pour into a lower plenum that is essentially empty or an explosion produced by water pouring onto a lower plenum debris bed.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.095 (Response Revision 1)

Question:

Why is the probability of random ignition assumed to be 1 during the intermediate time? The basis for this question is that it is not conservative to assume that ignition is guaranteed in the intermediate time when it comes to global detonations. Presumably the steam content in the uniformly mixed gases inside the containment decreases as the PCCS is allowed to cool the containment shell. In the limit (dry mixtures), the concentration of hydrogen is about 14% in the AP1000, assuming 100% active cladding reaction, or about 19% assuming 100% reaction of all core zirconium. This mixture is becoming sufficiently sensitive to undergo a transition to detonation, especially if the entire containment is viewed as one confined compartment with a lot of clutter (individual compartments below the operating deck).

Westinghouse Response:

Detonation in the intermediate time frame is considered and quantified at node DTI on the containment event tree. The intermediate time frame essentially covers the time from the end of in-vessel hydrogen generation to 24 hours after core damage. Due to the PCS heat removal, natural circulation in the containment is strong, and the containment mixes quickly. For sequences in which the igniters are not functioning, a global burn of the well-mixed gases in the containment is assumed to occur with the probability of ignition of 1.0. The global burn is evaluated for the potential for flame acceleration.

The probability of DDT in the intermediate time frame is assumed to be the same as AP600 since the containment is well mixed and the increase in zirconium mass corresponds to the increase in the containment volume. In the AP600 PRA, mixture class probability distributions are developed considering uncertainties in the degree of zirconium oxidation and steam concentration over the time frame. The gas mixture composition is considered to be the same in all compartments, except the CMT room, where it is assumed to be dry air and hydrogen. This conservatism is introduced to overcome uncertainty in steam concentration below the operating deck due to stratification caused by the condensation on the PCS shell. Additionally, the CMT room has been assigned an unfavorable geometry classification.

Therefore, the AP1000 treatment of DDT in the intermediate time frame is conservative, and the assumption of guaranteed ignition facilitates the treatment.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

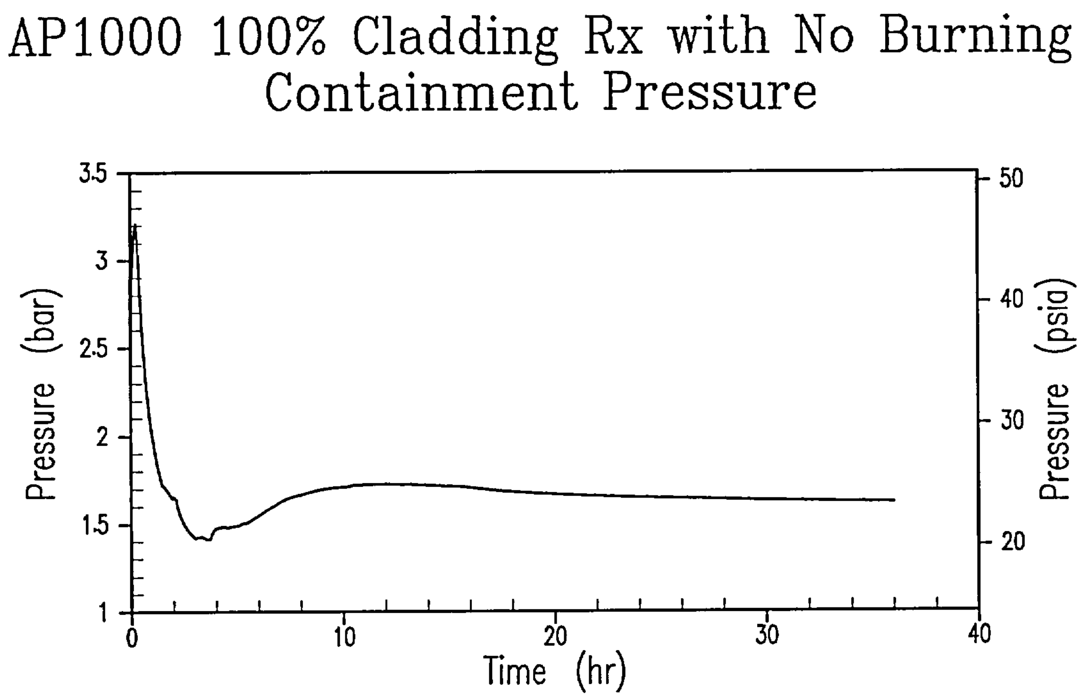
None

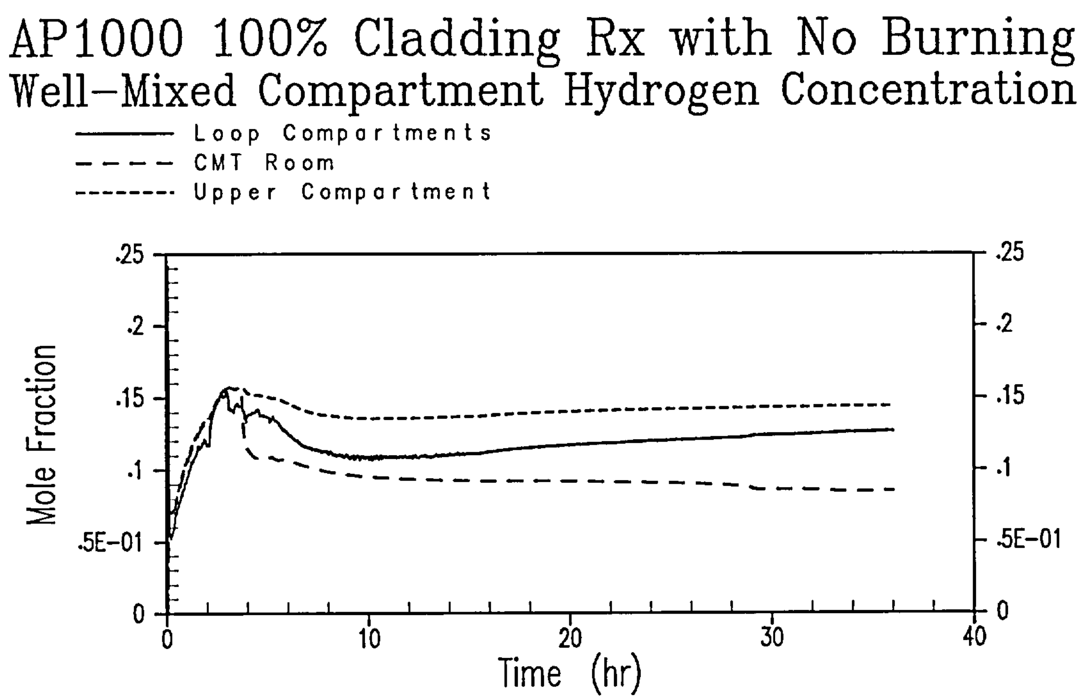
NRC Additional Comments:

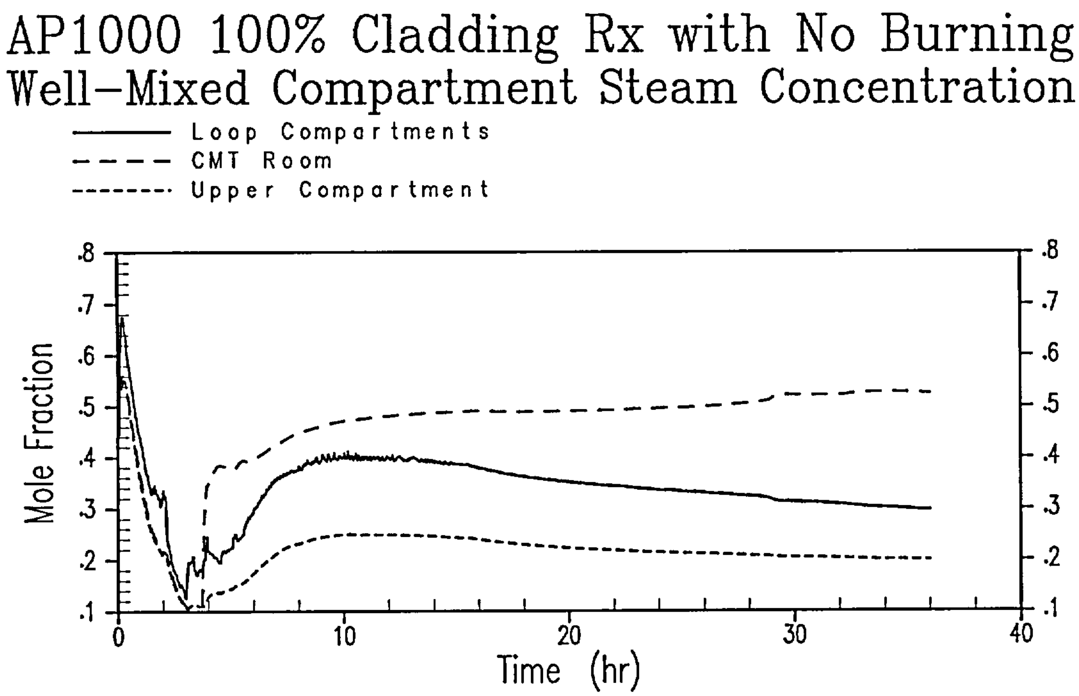
What are the mixture compositions within the AP1000 containment for a representative accident with 100% active cladding reaction throughout the entire sequence, including times beyond the intermediate time frame?

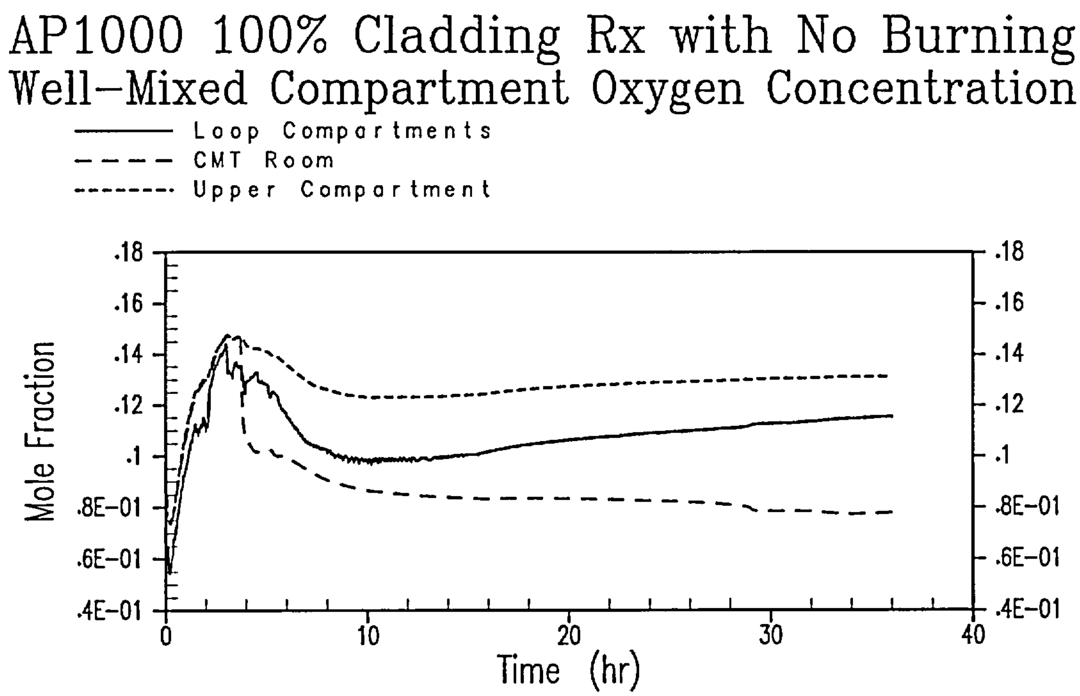
Westinghouse Additional Response:

The mixture composition results from a MAAP4 analysis that produces more than 100% cladding reaction is attached to this RAI. The accident is assumed to be a 3BR sequence with MAAP4 modeling parameters set to artificially maximize hydrogen production. This is the same sequence that was used to produce the global hydrogen burn environment in the equipment survivability analysis. However, there is no hydrogen burning assumed to occur in the analysis presented here, so the mixture composition is representative of 100% cladding reaction throughout the sequence. The analysis is run to 36 hours, which is beyond the intermediate time frame. Note that because hydrogen generation is completed in the early time frame and containment pressure is stable, mixture compositions do not change significantly from the intermediate to late time frames. Therefore, considering hydrogen combustion in the intermediate time frame is conservative with respect to the timing of containment failure and offsite dose.


AP1000 100% Cladding Rx with No Burning
Containment Pressure
3.5
3
2.5
2
1.5
1
Pressure (bar)
50
40
30
20
Pressure (psia)
0
10
20
30
40
Time (hr)


AP1000 100% Cladding Rx with No Burning
Well-Mixed Compartment Hydrogen Concentration
Loop Compartments
CMT Room
Upper Compartment
Mole Fraction
.25
.2
.15
.1
.5E-01
0
10
20
30
40
Time (hr)


AP1000 100% Cladding Rx with No Burning
Well-Mixed Compartment Steam Concentration
Loop Compartments
CMT Room
Upper Compartment
Mole Fraction
Time (hr)
.8
.7
.6
.5
.4
.3
.2
.1
0
10
20
30
40



AP1000 100% Cladding Rx with No Burning
Well-Mixed Compartment Oxygen Concentration
Loop Compartments
CMT Room
Upper Compartment
Mole Fraction
.18
.16
.14
.12
.1
.8E-01
.6E-01
.4E-01
0
10
20
30
40
Time (hr)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.096 (Response Revision 1)

Question:

What would be the safety margin basis for containment performance if the uncertainty in the range of steam inerting concentrations was used? The safety margin is less than 1 psi when hydrogen produced from 100% active cladding reaction is mixed with air saturated with 55% steam. However, there is uncertainty in the steam-inerting limits, as measurements have ranged from 49%-63% (M. G. Zabetakis, "Research on the Combustion and Explosion Hazards of Hydrogen-Water Vapor-Air Mixtures," AECU-3327, U. S. Atomic Energy Commission, September 1956.)

Westinghouse Response:

The safety margin basis is a beyond design basis calculation, and it is appropriate to use a best estimate value for the maximum steam concentration. Higher steam concentrations reduced the hydrogen and oxygen concentration to values below the lower bounds for globally flammable mixtures. The safety margin basis calculation as presented contains adequate conservatism in assuming 100% cladding reaction, failure of hydrogen control, and global burning that occurs at a high pressure that is highly unlikely at the time when hydrogen is present in the containment. Additional conservatism is unnecessary.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

Please provide a detailed sample calculation for the problem of solving the AICC pressure (equations shown in Section 41.9.2 (Revision 0)). It is presumed that this is the procedure used to produce the values in Table 41-4 and the basis for the values reported in Section 41.11. For example, using Equation 41-2 and the values given below the equation, it is not possible to obtain the same values for gas masses shown in Table 41-4. Furthermore, Equation 41-6 lists four gas constituents yet Table 41-4 lists five. If one uses the values provided in Section 41.9.2, one would get estimates of the AICC pressure that exceed the ASME service level C stress intensity limit of 91 psig.

W to clarify response.

Westinghouse Additional Response:

In the calculation for the safety margin basis, the initial conditions reflect the initial masses of oxygen, nitrogen and carbon dioxide in the containment assuming initially dry conditions, then imposing 55% steam concentration as the peak LOCA pressure at which a burn could occur. The dry volume, pressure and temperature are:

P_0 = initial pressure = 14.7 psia = 1.01 bar = 1.01×10^5 Pa
T_0 = initial temperature = 120°F = 322°K
V = containment volume = 2.07×10^6 ft^3 = 58622.4 m^3
R = the ideal gas constant = 8314 J/kg-mole/K

The number of moles of air is found from the ideal gas law (equation 41-2 in the text).

n_{air} = number of moles of air = $\frac{P_0 V}{R T_0}$ = 2211.7 kg-moles

The initial masses of the constitutes of air are:
M_{N2} = mass of nitrogen = 0.76 * 2211.7 * 28 = 47065 kg nitrogen
M_{O2} = mass of oxygen = 0.20 * 2211.7 * 32 = 14155 kg oxygen
M_{CO2} = mass of carbon dioxide = 0.04 * 2211.7 * 44 = 3893 kg carbon dioxide

The initial mass of hydrogen corresponds to 100% active cladding reaction.
M_{H2} = mass of hydrogen = 788 kg

Initial Masses and Moles of Dry Air and Hydrogen		
Constitute	Mass	kg-moles
Nitrogen	47065 kg	1681 kg-moles
Oxygen	14155 kg	442.3 kg-moles
Carbon Dioxide	3893 kg	88.5 kg-moles
Hydrogen	788 kg	394 kg-moles
	Total Number of Moles	2606 kg-moles

Number of moles of steam for 55% concentration = 2606/0.45 – 2606 = 3185 kg-moles
Initial mass of steam in containment = 57330 kg.

The specific volume of the steam, 1.023 m^3/kg, is known from the ratio of the total volume to the mass of steam. The initial gas temperature of 388.3°K is found from the saturation temperature at the partial pressure of the steam, which is 1.72 bar.

The partial pressure of the non-condensable gases is found using the ideal gas law:
$PP_{NC} = nRT/V = 2606 * 8314 * 388.3 / 58622.4 = 1.44$ bar

The total pressure is $PP_{st} + PP_{NC} = 1.72 + 1.44 = 3.16$ bar.

The constant volume specific heats at the initial and final temperatures are found using the equations for Cp_o from Table A.9 in reference 1. The Cp_0 is converted to Cv by noting that: $Cp_X + Cv_X = R/mw_X$ for each particular constitute, X.

The specific heats are summarized in Table 41-4 of the PRA report.

The final temperature is calculated using equation 41-6, and noting that CO_2 was inadvertently omitted from the list of constitutes in equation 41-6. The pressure is calculated with equation 41-7, the ideal gas law.

		mass (kg)	spec ht (J/kg-k)	final kg-moles	
N2	initial	47065	745		
	final	47065	851.4	1681	
CO2	initial	3893	739.8		
	final	3893	1011.2	88.5	
Steam	initial	57330	1432.9		
	final	64371	1762.8	3576	
Oxygen	initial	14155	676.5		
	final	7851	810.7	245.3	
Hydrogen	initial	788	10444		
	final	0		5591	total

Qb = 9.53E+10 Joules

Tf = 908.7 K
Pf = 7.29 bar   91.0 psig

The hand calculation here verifies the calculation presented in Table 41-4.