Westinghouse

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

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, D.C. 20555

Direct tel: 412-374-5355
Direct fax: 412-374-5456
e-mail: corletmm@westinghouse.com

Your ref: Docket No. 52-006
Our ref: DCP/NRC1617

September 8, 2003

SUBJECT: Transmittal of Responses to AP1000 DSER Open Items

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

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

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

D063

September 8, 2003

The Westinghouse Electric Company Application for Withholding and Affidavit are also attached to 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.

Correspondence with respect to the application for withholding should reference AW-03-1696, 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 at 412-374-5355 if you have any questions concerning 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-03-1696.

/Attachments

1. List of the AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses transmitted with letter DCP/NRC1617
2. Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated September 8, 2003
3. Non-Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated September 8, 2003

DCP/NRC1617
Docket No. 52-006

September 8, 2003

Enclosure 1

Westinghouse Electric Company
Application for Withholding and Affidavit

Westinghouse

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

September 8, 2003

AW-03-1696

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

ATTENTION: Mr. John Segala

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

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

Dear Mr. Segala:

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

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject documents. In conformance with 10 CFR Section 2.790, Affidavit AW-03-1696 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-03-1696 and should be addressed to the undersigned.

Very truly yours,

M. M. Corletti

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

/Enclosures

3141alf.doc

AW-03-1696

COMMONWEALTH OF PENNSYLVANIA:

ss

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 8th day
of September, 2003

Lorraine M. Piplica
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

3141alf.doc

(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.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in Attachment 2 as Proprietary Class 2 in the Westinghouse Electric Co., LLC document: (1) "AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Response."

This information is being transmitted by Westinghouse's letter and Application for Withholding Proprietary Information from Public Disclosure, being transmitted by Westinghouse Electric Company (W letter AW-03-1696) 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 Control Document," analysis on a plant specific basis

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

Further this information has substantial commercial value as follows:

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

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

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar methodologies and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

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

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

Further the deponent sayeth not.

September 8, 2003

Attachment 1

List of

Proprietary and Non-Proprietary Responses

Table 1 "List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1617"	
5.2.3-2 5.2.3-3 6.2.1.8.3.3-1 Rev 2 14.2-1aa Rev 1 14.2.1bb Rev 1 *15.2.7-1P Rev 1 15.2.7-1 Rev 1 *Proprietary	

DCP/NRC1617
Docket No. 52-006

September 8, 2003

Attachment 3

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

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 5.2.3-2

Original RAI Number(s): None

Summary of Issue:

Alloy 52/152 materials are known to be difficult to weld. Address what examinations have been given to the adequacy of quality assurance (QA) criteria for the Alloy 51/152 weldments that will be used to connect stainless steel piping to the ferritic pressure vessel. Address whether the QA criteria are commensurate with the risk associated with weldment failure.

Westinghouse Response:

The "difficulties of welding" ascribed to Alloys 52 and 152 refers to the fact that large section multipass welds made with these materials have been observed to contain hot cracks. Such hot cracks are controlled by the application of proper weld technique, but are difficult to totally eliminate. In addition, welding with the shielded metal arc process using the Alloy 152 coated electrode requires frequent back-chipping and grinding to eliminate "floaters", which are small (Al and Ti) oxide inclusions.

The Quality Assurance criteria for these weldments are essentially the same as those previously and currently applied to weldments of Alloys 82 and 182. The welds must pass the required ASME Boiler & Pressure Vessel (B&PV) Code Section III requirements - i.e., for dye penetrant inspections, no reportable indications (that is, no linear indications) are permitted on the final weld surface. If they are found to be present, they must be removed and the welds repaired as specified by the ASME B&PV Code. This results in no surface cracks in the weld. The presence of minor subsurface hot cracks is of no consequence for at least three reasons: (1) they are not in contact with the primary water; (2) the excellent corrosion resistance of Alloys 52 and 152 will preclude the occurrence of stress corrosion cracking at the water-weld metal interface (these materials have never been observed to experience environmental degradation in primary water): and (3) the very small subsurface hot cracks have not been found to serve as crack initiation or propagation sites. This Quality Assurance criteria is intended to minimize the risk of weldment failure.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 5.2.3-3

Original RAI Number(s): None

Summary of Issue:

The high-chromium nickel-base alloys (e.g., Alloy 690/52/152, as well as 82/182) may be susceptible to a significantly lowered fracture toughness if they have been exposed to high temperature hydrogenated water and then stressed at lower temperature (e.g., < 120°C). This is a known phenomenon and may be of significance during a thermal shock event (i.e., during an accident scenario when there is ingress of large amounts of cold water into the primary system). Address whether this phenomenon could result in the failure of the nozzles between the pressure vessel and main recirculation or direct vessel injection (DVI) piping. If such a failure occurred, what are the consequences?

Westinghouse Response:

Background

The scenario reflected in the open item is that, since it is possible that small subsurface weld flaws such as hot cracks or floaters/stringers may be present in heavy section multipass welds of Alloys 52 and 152, might these flaws serve as sites for the initiation and subsequent growth of cracks. There have been a reported decrease of toughness of these welds in low temperature hydrogenated water environments. Could such degradation occur for an accident scenario in with large amounts of cold water rapidly lowering the temperature before hydrogen is removed from the metal and the reactor coolant?

Under normal plant shutdown conditions such a scenario is not possible since hydrogen is removed by a combination of mechanical and chemical degassing before the water temperature gets below about 180°F.

Essentially all published research on the phenomenon referred to as low temperature crack propagation (LTCP) has been the result of research conducted by Mills and co-workers at Bettis (cf., Refs. 1-4).

Literature

The published research was reviewed and an abbreviated summary of relevant results with the authors' conclusions follow.

The materials that have been studied for susceptibility to the LTCP phenomenon include Alloy X-750, Alloys 600 and 690, and the weld metal Alloys EN82H and EN52. All but Alloy 600 have been found susceptible to varying degrees. The reason for the exception of Alloy 600 is not clear. The general ranking of the alloys found susceptible is (most to least susceptible): Alloy X-750, Alloy EN82H, Alloy EN52, Alloy 690. Each of these alloys exhibits extremely high fracture resistance in air and high temperature water, irrespective of the presence of hydrogen.

With regard to the present issue, the research reported in Ref. 4 is most relevant. Deeply precracked compact tension specimens of Alloys EN82H, EN52, 690 and 600 were tested at 54°C (130°F), 93°C (200°C), 149°C (300°C) and 338°C (640°F) water containing concentrations of dissolved hydrogen ranging from 15 to 150 cm^3 (STP) H_2/kg H_2O. Limited testing was also performed in 54°C (130°F) and 338°C (640°F) air.

The specimens were tested to fracture using a displacement rate of 4 MPa√m/h at 54°-149°C and 0.4 to 2 MPa√m/h at 338°C to assure sufficient time for environmental cracking. Multiple-specimen heat-treat and single-specimen normalization J-curve test procedures were used to establish J_{IC} and tearing modulus values (Refs. 1, 5). J_{IC} is the fracture toughness at the onset of cracking and the tearing modulus is a dimensionless measure of a material's resistance to cracking after J_{IC} is exceeded (see Ref. 4 for definition of terms).

For the tests in air, all materials exhibited extremely high values of J_{IC} and T. However, in 54° and 93°C water the values were severely reduced. The following table summarizes the J_{IC} and tearing modulus results for the 54°C tests in hydrogenated water.

Material	J_{IC} at 54°C, kJ/m²/Tearing Modulus		
	15 cm^3 H_2/kg H_2O	50 cm^3 H_2/kg H_2O	150 cm^3 H_2/kg H_2O
Alloy 690 Heat A	150/128	90/72	75/38
Alloy 690 Heat B	120/63	95/58	25/24
EN52 Heat B1	No test	No test	~ 50/38
EN52 Heats C1, C2	100/53	80/59	20/4
EN82H Heat C1	8/3	No test	10/5
Alloy 600 Heat A	285/232	No test	285/232

The authors conclude from these data that, in view of the reasonably high tearing moduli for Alloy 690 (24 to 38 at the highest hydrogen concentration), LTCP is not a concern for that alloy.

For the embrittled welds, the authors calculate equivalent critical stress intensity factors (K_{JC}) from the experimental J_{IC} values, and find good agreement with the maximum stress intensity values (K_{Pmax}) observed in previous rising-load tests. [A thorough description of the testing and data interpretation for the rising load tests is provided in Ref. 3.] This agreement suggests that cracking initiates near maximum load under nearly linear-elastic conditions; as a result, lower-bound K_{Pmax} values can be used as a measure of LTCP resistance.

The resulting lower-bound K_{Pmax} values for EN82H and EN52 are 40 and 53 MPa√m, respectively. [Observe that these are not 'low' stress intensities.]

Additional aspects of the Ref. 4 test program included consideration of the displacement rate and the effect of crack geometry. For displacement rates greater than 1000 MPa√m/h (approximately 12 mm/h per the authors for this geometry), LTCP resistance of both weld metals improves significantly, leading the authors to conclude that this phenomenon "is not an issue for welded components subjected to rapid transients."

The authors also observed (in Ref. 4 and other publications) that intergranular LTCP does not occur from as-machined notches or at free surfaces in the absence of a sharp corrosion-induced crack (or perhaps even a small ductile tear). This is interpreted in terms of the requirement that hydrogen is concentrated at the crack tip, and this requires solid-state diffusion of hydrogen.

Other research (Refs. 1 – 3) by the same authors indicates the following:

- LTCP is due to hydrogen embrittlement of the grain boundary regions adjacent to and immediately ahead of the advancing crack,
- The phenomenon appears only under conditions of rising loads – i.e., not under constant load conditions,
- For Alloy 52, the critical stress intensity must be greater than 50 MPa√m, and
- The effective displacement rate must be quite low.

The latter point has been interpreted as supporting the judgment that solid-state diffusion of hydrogen is rate-controlling. This was further supported by calculations and measurements of the activation energies for LTCP crack growth rates (11.3 kcal/mole) and for hydrogen diffusion at this temperature (11.5 kcal/mole). This is also consistent with the observation that LTCP does not occur in the presence of a notch since the diffusion distance to the peak stress location ahead of a notch is much greater than that for a sharp (SCC) crack.

Relevance to Alloy 52/152 Welds in AP1000

Alloys 52 and 152 will be used for large section, multipass welds such as reactor vessel nozzle-to-safe-end welds in AP1000. It is possible that subsurface flaws such as small hot cracks or oxide "floaters" may be present.

However, based on the preceding review of the literature, there appears to be no technical basis to argue such welded structures will be susceptible to low temperature crack propagation under normal or accident-driven conditions. The basis for this conclusion rests on the conditions under which LTCP has been observed to occur in these materials. These conditions include:

- high loads that are applied slowly, increase with time, and are capable of producing high stress intensities, and
- the presence of sharp intergranular (or ductile) cracks.

Hot cracks or subsurface welding defects are unlikely to be characterized by a sharp, "crack-tip" geometry such that high hydrogen concentrations will develop and promote crack advance. Also, the relatively rapidly applied high loads that may be associated with a surge of low temperature water under accident conditions are not consistent with LTCP occurrence. Moreover, the extremely high resistance of these high chromium alloys to stress corrosion crack initiation in primary water makes it extremely unlikely that in-service cracking will occur; hence, a low probability that a surface crack will exist to serve as the propagation site.

References

1. C. M. Brown and W. J. Mills, "Fracture Toughness of Alloy 690 and EN52 Weld in Air and Water," Bettis Atomic Power Laboratory Report, B-T-3265, June 1999.
2. W. J. Mills and C. M. Brown, "Fracture Toughness of Alloy 600 and an EN82H Weld in Air and Water," Metallurgical and Materials Transactions A, 32A (May 2001) 1161-1174.
3. W. J. Mills, M. R. Lebo and J. J. Kearns, "Hydrogen Embrittlement, Grain Boundary Segregation, and Stress corrosion Cracking of Alloy X-750 in Low- and High-Temperature Water," Metallurgical and Materials Transactions A, 30A (June 1999) 1579-1596.
4. W. J. Mills and C. M. Brown, "Fracture Behavior of Nickel-based Alloys in Water," *Ninth International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors*, TMS, 1999.
5. W. C. Porr and W. J. Mills, "Application of the Normalization Data Analysis Technique for Single Specimen R-Curve Determination," Bettis Atomic Power Laboratory Report B-T-3269, February 1999.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 6.2.1.8.3-1 (Revision 2)

Original RAI Number(s): 650.001

Summary of Issue:

The water level in containment following a LOCA would be sufficiently high that DCD Tier 2 Section 3.4.1.2.2.1 states that inventory from the containment pool would ". . . flow back into the RCS via the break location . . . ." In light of this statement, the staff issued RAI 650.001 to request additional information concerning the potential for entrained debris to cause blockage at flow restrictions within the RCS once flow begins entering through the break location after flood-up (i.e., bypassing the recirculation screens). In a letter dated February 21, 2003, the applicant responded to RAI 650.001 by submitting an analysis which concluded that RMI debris is incapable of causing such blockage. Although the applicant's response partially addressed the staff's RAI, it was not complete because it did not address the potential for other sources of debris, such as fibrous debris and floatable debris, to enter the RCS through the break location and block requisite core cooling flowpaths. Pending the complete resolution of this concern, the staff considers debris blockage in the RCS to be DSER Open Item 6.2.1.8.3-1.

Westinghouse Response:

Westinghouse revised its response to RAI 650.001 in order to address the NRC concerns as discussed in our teleconference on April 3, 2003. The revised RAI response was submitted to the NRC on April 24, 2003 in letter DCP/NRC1580. Based on discussions with the NRC after the issuance of the DSER, it was agreed that this response satisfactorily addressed the NRC concerns except for the calculated debris pressure loss. Westinghouse agreed to revise the calculation of the pressure loss across a debris bed located in the core and to perform additional sensitivity studies on particulate characteristics. The revised calculation and sensitivity studies are~~is~~ based on the following:

1. A total of 500 lb of resident debris (fiber and particles) is assumed to be located inside containment.
2. This debris is assumed to be neutrally buoyant (both fibers and particles) such that they are easily transported with flow.
3. The resident debris is distributed around the containment in proportion to the floor areas.
4. If a floor area sees flow either from LOCA blowdown, ADS venting or containment recirculation, then debris assigned to that floor area is assumed to be transported to a screen.
5. If a floor area does not see flow (whether it floods or not) then none of the debris assigned to that floor area is assumed to be transported.
6. The head losses across the screens will be calculated using the "BLOCKAGE" code. The resident debris fiber material is assumed to be represented by NUKON.

7. Sensitivity studies will be performed with variations in ~~both~~ the amount of debris transported to the screens, ~~and~~ in the mass ratio of fiber versus particulate debris and in the types of particulates assumed in the resident debris.

Based on these assumptions, methods and approaches, the head loss analysis performed for RAI 650.001 (debris in the core) was revised.

Resident Fibrous and Particle Debris:

A potential source of debris is resident fiber and particles inside containment. Such debris might be close enough to the density of water that it would stay suspended in the containment water long enough that it could be transported to containment recirculation screens and possibly also into the RCS through a break that becomes flooded.

DSER open item response 2.1.8.3-3 R1 discusses the amount of such debris that might exist in the containment. It describes an appropriate method to determine the amount of debris that might be transported. It also describes an appropriate method using the BLOCKAGE code to calculate the resulting pressure drop if this debris is transported to a containment recirculation screen. That same method has been applied to a situation with a break location that becomes flooded and could allow some of this debris to enter the RCS. Key assumptions made in this evaluation include:

- A total of 500 lb of resident debris is located in the containment (DSER OI 2.1.8.3-3 R1). The base case assumes that this debris is divided 50/50 between fibers and particles. Also, as described below, sensitivity studies are also performed assuming a range of particulate to fiber ratios.

- The debris is distributed around the containment in proportion to the floor areas. As discussed in DSER OI 6.2.1.8.3-3 R1, not all of this debris will be transported because some floor areas will not see flow during a LOCA.

- The limiting break location with respect to maximizing the debris that might enter the RCS has been determined to be a DVI break in a loop compartment. Such a break will result in none of the operating deck and only a portion of the CMT room floor (< 67%) seeing flow. As a result, less than 250 lb of resident debris will be transported.

- The debris deposited on any screen is assumed to be based on the flow split about containment. As noted above, for the DVI break in a loop compartment, less than 250 lb of resident containment debris is available for transport. Of this amount of debris, about 100 lb of debris will be transported to the IRWST screens. The remainder (150 lb) will be transported to the recirculation screens and to the RCS via the break. This 150 lb is further divided in the proportion of the relative flows as described below.

- Conservative analyses have shown that 60% of the total flow to the core is through the break and 40% through the recirculation screens. Assuming the debris transport is proportional to the flow, 60% of the resident debris will enter the RCS through the break

(90 lb). The other 40% (60 lb) would to be trapped on the two recirculation screens. These debris amounts are based on the relative flows through the break and through the PXS recirc lines as shown on DCD figures 15.6.5.4C-13 and -14 after 7000 sec. Although the flow through the break into the RCS starts earlier than through the PXS recirc lines, it would take many hours to transport all of the debris to the RCS / recirc screens. For example, the total water mass in the containment floodup areas is about 5,236,000 lb. At a recirc flow of 180 lb/sec it would take about 10 hours for all of this water to flow through the RCS. The situation for the recirc screens is much less limiting than that discussed in DSER OI response 2.1.8.3-3 R1, so that the resulting it is not discussed in this RAI.

- The first location where debris may be trapped in the RCS is on the bottom nozzle of the fuel assembly. Each nozzle has 632 flow holes that are 0.19 in inside diameter. These holes are spaced such that debris would accumulate across the whole nozzle area except the outside edge where there are no holes. The area that could accumulate debris is more than 66 ft^2 considering all of the fuel assemblies. Another location where debris could be trapped is in the P-Grid, which is located just above the bottom nozzle. The area where debris could accumulate is defined as the fuel assembly area less the area taken by the fuel rods and thimbles for shutdown rods and I&C. The minimum flow area through this part of the core is 41.55 ft2. The smaller area (around the P-Grid) is assumed for the purposes of calculating the pressure loss.

- The flow rate through the core is assumed to be 180 lb/sec. This flow is based on the maximum injection flows through both DVI lines as shown on DCD figures 15.6.5.4C-13 and -14 after 7000 sec.

- Using the core inlet temperature from COBRA-TRAC calculations for this event (~240 F), the volumetric flow rate would be 1370 gpm.

- At this flow rate, the screen face velocity with this flow is 0.073 ft/sec.

- With the above amounts of debris and flow rates, the pressure loss across the debris is calculated by the BLOCKAGE code to be less than 0.39 psi. A summary of BLOCKAGE Code input and resulting output for the base case are shown on ~~the~~ table 6.2.1.8.3_1-1 that follows. Refer to DSER OI response to 2.1.8.3-3 R1 for additional discussion on the use of the BLOCKAGE code.

- In addition to the base calculation, sensitivity studies were performed on the amount of debris transported and the mass ratio of fiber to particulate debris.
  - Sensitivity calculations were performed varying the total mass of material from 80% (72lb.) to 120% (108lb.) of the base case (90 lb). This sensitivity addresses possible variability in the amount of debris available to transport.
  - Fiber to particulate mass ratios ranging from 30% fiber/ 70% particulate to 70% fiber/ 30% particulate were investigated for all three total mass cases. This sensitivity addresses the impact of fiber to particulate ratios different from the base case assumption of a 50/50 split.

The results of these sensitivity analyses are shown in ~~on the~~ table 6.2.1.8.3_1-1 that follows. The pressure drops for all the cases investigated ranged from 0.25 psid to 0.63 psid. For the range of masses and mass ratios investigated, the range of calculated pressure drop values was narrow and the trend of pressure drops within the range showed no unexpected results.

- **A second set of sensitivity studies was performed on the types of particulate debris assumed in the resident debris. Several different types of debris were modeled and the results compared to the base case debris type used in this analysis. The results of these sensitivity analyses are shown in table 6.2.1.8.3_1-2 that follows.**
  - **Blockage runs were made for alternate particle debris types; in the first set of alternate debris analysis the only change from the previous analysis was to increase the particle specific surface area from 20,000 to 50,000 ft2/ft3. This change creates a very conservative situation of a large specific surface area with a low specific gravity (1.1); both are drivers for a larger pressure drop. The results of these runs show that the pressure drop for the base case (50% fiber/50% particle) increases from 0.39 psi to 0.76 psi. The alternate case with more particles (30% fiber/70% particle) results in an higher pressure drop of 0.96 psi and the case with less particles (70% fiber/30% particle) results in a lower pressure drop of 0.74 psi.**
  - **Similar Blockage runs were made for particles with attributes of Analytical Test Problem Debris from NUREG/CR-6371, Reference 1, as shown in Table 4-3 of that report. The BLOCKAGE runs were made for the debris types of Paint, Junk, and Cal. Silicate with the following attributes:**
    - **Paint: Particle specific surface area = 50,000ft2/ft3; Particle fabricated density = 180lb/ft3; Particle Rubble Density = 45lb/ft3; Particle Material Density = 180lb/ft3.**
    - **Junk: Particle specific surface area = 900ft2/ft3; Particle fabricated density = 300lb/ft3; Particle Rubble Density = 95lb/ft3; Particle Material Density = 491lb/ft3.**
    - **Cal. Silicate: Particle specific surface area = 20,000ft2/ft3; Particle fabricated density = 90lb/ft3; Particle Rubble Density = 20lb/ft3; Particle Material Density = 110lb/ft3.**

  The results from these runs in Table 6.2.1.8.3_1-1, show that the pressure drops calculated for the particle characteristics used in the base case calculations are representative of the types of particles that may be present in the AP1000 containment. The resulting pressure drops for these three representative particulate types are very similar to the base case pressure drop values and are less than the pressure drop results from the arbitrary particle characteristic combination of 1.1 specific gravity and a 50,000 ft2/ft3 specific surface area.

Reference 1 contains characteristics for various types of particulate material that has been postulated to be present inside containment buildings of nuclear power plants. These types of particulate material are recommended for

consideration when a plant evaluates the susceptibility of ECCS sump performance degradation to the presence of resident fibers and materials inside containment. The studies performed by Westinghouse in the evaluation of the AP1000 as presented in this DSER Open Item response and related responses have included bounding material properties that are considered appropriate for the AP1000. Products of corrosion are not expected to play a role in the pressure drops considered in this analysis for the AP1000. Therefore, the pressure drops for the debris type of Sludge as defined in Reference 31 is not considered. These products of corrosions are included in Reference 3 for application to BWR containment buildings that contain carbon steel-lines suppression pools tanks that come in contact with oxygenated water. These corrosion products are not applicable to the AP1000 or other pressurized water reactor designs.

- The mechanism for driving flow through the core is the water level in the downcomer relative to the water/steam mixture level in the core region. In this case the downcomer water level is about 22 in below the top of the active fuel in the recirculation time frame (7000 sec), as shown in DCD figure 15.6.5.4C-1. This level is about 70 in below the DVI connection to the reactor vessel. The injection from the DVI lines would not be affected by the downcomer water level as long as the level is below the DVI connection. Therefore in case there is an additional pressure loss of 0.39 psi across the core, the downcomer water level would increase by about 12 in so that the flow through the core is maintained. The water level in the downcomer would still be 58 in below the DVI connection.

  Even if the pressure drop was 1.0 psi across the debris, the downcomer water level would increase by 30 in. (instead of 12 in.) and would still be well below (40 in.) the DVI connection. The flow through the core would be unaffected. This pressure drop bounds the pressure drop calculated assuming a high percentage of particles (70%) and the arbitrary particle characteristics of 1.1 specific gravity and a 50,000 ft2/ft3 specific surface area.

In summary, the bounding pressure loss through a conservatively large amount of resident debris that might deposit ~~on the lower core support plate or~~ in the core would not reduce the flow to the core. In order to provide additional confidence that the above calculated pressure drops are bounding, a COL item will be added to verify that potential resident particles have an average specific surface area $\leq$ 50,000 ft2/ft3 and an average specific gravity $\geq$ 1.1. The determination of these characteristics will be based on sample measurements from operating plants, such as the research planned by the NRC to characterize latent debris (Reference 2).

References:

1. NUREG/CR-6371, "BLOCKAGE 2.5 Reference Manual", December 1996.

2. "Recent Results and Future GSI-191 Research" (Presentation at NEI PWR Sump Performance Workshop, Baltimore, Maryland, 7/30-31/03), B. Letellier, Los Alamos National Laboratory

Table 6.2.1.8.3_1-1. Core Pressure Drop									
	Mass Debris (lbm)	Percent Fiber	Percent Particulate	Mass Fiber (lbm)	Mass Particulate (lbm)	Thickness (in)	Pressure Drop (ft-water)	Pressure Drop (psf)	Blockage Case Name
	90	30%	70%	27	63	3.26	0.72	0.31	APCO_301
	90	40%	60%	36	54	4.34	0.79	0.34	APCO_302
Base Case	90	50%	50%	45	45	5.43	0.91	0.39	APCO_303
	90	60%	40%	54	36	6.51	1.04	0.45	APCO_304
	90	70%	30%	63	27	7.59	1.21	0.52	APCO_305
80% of Total Debris	72	30%	70%	22	50	2.61	0.57	0.25	APCO_311
	72	40%	60%	29	43	3.47	0.63	0.27	APCO_312
	72	50%	50%	36	36	4.30	0.72	0.31	APCO_313
	72	60%	40%	43	29	5.21	0.83	0.36	APCO_314
	72	70%	30%	50	22	6.08	0.97	0.42	APCO_315
120% of Total Debris	108	30%	70%	32	76	3.90	0.86	0.37	APCO_306
	108	40%	60%	43	65	5.24	0.96	0.42	APCO_307
	108	50%	50%	54	54	6.52	1.09	0.47	APCO_308
	108	60%	40%	65	43	7.82	1.25	0.54	APCO_309
	108	70%	30%	76	32	9.12	1.45	0.63	APCO_310

INPUT TO BLOCKAGE CODE			
Value	Parameter	Description	Note
AP1000 Calculation Fiber and Particulate Debris Parameters			
0.986	ε_f	Pure fiber bed porosity	
175	ρ_f	Fiber density (lb_m/ft^3) also Material density	
2.4	c_0	Fabricated Fiber Density	
68.64	ρ_p	Particle density (lb_m/ft^3)	
1.71E+05	Sv	Specific (volumetric) surface area (ft^2/ft^3)	
AP1000 Core Pressure Drop			
1370	685	Flow of Water though Recirc. Screen (GPM)	
200		Temperature of water at screen (deg. F.)	
41.55		Core Flow Area (ft2)	
90		Mass of Total Debris (lbm) Base Case	

Core Pressure Drop									
	Mass Debris (lbm)	Percent Fiber	Percent Particulate	Mass Fiber (lbm)	Mass Particulate (lbm)	Thickness (in)	Pressure Drop (ft-water)	Pressure Drop (psi)	Blockage Case Name
100% of Total Debris	90								
	90	30%	70%	27	63	3.26	0.72	0.31	APCO_301
	90	40%	60%	36	54	4.34	0.79	0.34	APCO_302
Base Case	90	50%	50%	45	45	5.43	0.91	0.39	APCO_303
	90	60%	40%	54	36	6.51	1.04	0.45	APCO_304
	90	70%	30%	63	27	7.59	1.21	0.52	APCO_305
80% of Total Debris	72	30%	70%	22	50	2.61	0.57	0.25	APCO_311
	72	40%	60%	29	43	3.47	0.63	0.27	APCO_312
	72	50%	50%	36	36	4.30	0.72	0.31	APCO_313
	72	60%	40%	43	29	5.21	0.83	0.36	APCO_314
	72	70%	30%	50	22	6.08	0.97	0.42	APCO_315
120% of Total Debris	108	30%	70%	32	76	3.90	0.86	0.37	APCO_306
	108	40%	60%	43	65	5.24	0.96	0.42	APCO_307
	108	50%	50%	54	54	6.52	1.09	0.47	APCO_308
	108	60%	40%	65	43	7.82	1.25	0.54	APCO_309
	108	70%	30%	76	32	9.12	1.45	0.63	APCO_310

		INPUT TO BLOCKAGE CODE	
Value	Parameter	Description	Note
AP1000 Calculation Fiber and Particulate Debris Parameters			
0.986	ε_f	Pure fiber bed porosity	NUKON, Reference 2
175	ρ_f	Fiber density (lb_m/ft^3) also Material density	NUKON, Reference 2
2.4	c_0	Fabricated Fiber Density	NUKON, Reference 2
68.64	ρ_p	Particle density (lb_m/ft^3)	Specific Gravity of 1.1
1.71E+05	Sv	Specific (volumetric) surface area (ft^2/ft^3)	NUKON, Reference 2
AP1000 Core Pressue Drop			
1370	685	Flow of Water though Recirc. Screen (GPM)	
200		Temperature of water at screen (deg. F.)	
41.55		Core Flow Area (ft2)	Reference 4
90		Mass of Total Debris (lbm) Base Case	

3

Table 6.2.1.8.3_1-2. Sensitivity Studies: Blockage Runs on Varying Attributes of the Particulate Debris

	Mass Debris (lbm)	Percent Fiber	Percent Particulate	Mass Fiber (lbm)	Mass Particulate (lbm)	Particle Sv (ft2/ft3)	Particle Fab. Density (lb/ft3)	Particle Material Density (lb/ft3)	Thickness (in)	Pressure Drop (ft-water)	Pressure Drop (psi)	Blockage Case Name
	90	30%	70%	27	63	20,000	68.64	68.64	3.26	0.72	0.31	APCO_301
	90	40%	60%	36	54	20,000	68.64	68.64	4.34	0.79	0.34	APCO_302
Base Case	90	50%	50%	45	45	20,000	68.64	68.64	5.43	0.91	0.39	APCO_303
	90	60%	40%	54	36	20,000	68.64	68.64	6.51	1.04	0.45	APCO_304
	90	70%	30%	63	27	20,000	68.64	68.64	7.59	1.21	0.52	APCO_305
Part. SG=1.1, Sv=50,000	90	30%	70%	27	63	50,000	68.64	68.64	3.26	2.21	0.96	APCOe301
	90	50%	50%	45	45	50,000	68.64	68.64	5.43	1.76	0.76	APCOe303
	90	70%	30%	63	27	50,000	68.64	68.64	7.59	1.71	0.74	APCOe305
Paint (Note 1)	90	30%	70%	27	63	50,000	180.00	180.00	3.26	0.98	0.42	APCOf301
	90	50%	50%	45	45	50,000	180.00	180.00	5.43	1.19	0.52	APCOf303
	90	70%	30%	63	27	50,000	180.00	180.00	7.59	1.45	0.63	APCOf305
Junk (Note 1)	90	30%	70%	27	63	900	300.00	491.00	3.26	0.43	0.19	APCOg301
	90	50%	50%	45	45	900	300.00	491.00	5.43	0.83	0.36	APCOg303
	90	70%	30%	63	27	900	300.00	491.00	7.59	1.25	0.54	APCOg305
Cal. Silicate (Note1)"	90	30%	70%	27	63	20,000	90.00	110.00	3.26	0.59	0.26	APCOh301
	90	50%	50%	45	45	20,000	90.00	110.00	5.43	0.87	0.38	APCOh303
	90	70%	30%	63	27	20,000	90.00	110.00	7.59	1.22	0.53	APCOh305

INPUT TO BLOCKAGE CODE

Value	Parameter	Description	Note
AP1000 Calculation Fiber and Particulate Debris Parameters			
0.986	ε_f	Pure fiber bed porosity	
175	ρ_f	Fiber density (lbm/ft3) also Material density	
2.4	c_0	Fabricated Fiber Density	
68.64	ρ_p	Particle density (lbm/ft3)	
1.71E+05	S_v	Specific (volumetric) surface area (ft2/ft3)	
AP1000 Core Pressure Drop			
1370		Flow of Water though Core (GPM)	
200		Temperature of water at screen (deg. F.)	
41.55		Core Flow Area (ft2)	
Note 1		Debris Type Attributes from NUREG/CR-6371 Table 4-3, Reference 1	

Design Control Document (DCD) Revision:

~~None~~ DCD Table 1.8-2 will be revised as follows:

Table 1.8-2 (Sheet 3 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
4.3-1	Changes to Reference Reactor Design	4.3.4
4.4-1	Changes to Reference Reactor Design	4.4.7
5.2-1	ASME Code and Addenda	5.2.6.1
5.2-2	Plant Specific Inspection Program	5.2.6.2
5.3-1	Reactor Vessel Pressure - Temperature Limit Curves	5.3.6.1
5.3-2	Reactor Vessel Materials Surveillance Program	5.3.6.2
5.3-3	Surveillance Capsule Lead Factor and Azimuthal Location Confirmation	5.3.6.3
5.3-4	Reactor Vessel Materials Properties Verification	5.3.6.4
5.3-5	Reactor Vessel Insulation	5.3.6.5
5.4-1	Steam Generator Tube Integrity	5.4.15
6.1-1	Procedure Review for Austenitic Stainless Steels	6.1.3.1
6.1-2	Coating Program	6.1.3.2
6.2-1	Containment Leak Rate Testing	6.2.6
6.3-1	Containment Cleanliness Program	6.3.8.1
6.3-2	Verification of Containment Resident Particulate Debris Characteristics	6.3.8.2
6.4-1	Local Toxic Gas Services and Monitoring	6.4.7
6.4-2	Procedures for Training for Control Room Habitability	6.4.7
6.4-3	Main Control Room Inleakage Test Frequency	6.4.7

DCD section 6.3.8 will be revised as follows:

6.3.8 Combined License Information

6.3.8.1 Containment Cleanliness Program

The Combined License applicants referencing the AP1000 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages.

6.3.8.2 Verification of Containment Resident Particulate Debris Characteristics

The Combined License applicants referencing the AP1000 will determine that resident particles that could be present considering the plant location and the containment cleanliness program, have an average specific surface area ≤ 50,000 ft2/ft3 and an average specific gravity ≥ 1.1. The determination of these characteristics will be based on sample measurements from operating plants. (Terry – can we name the ongoing program here?) If these characteristics are not satisfied, then a determination will be made that the resident debris particle characteristics, when considering the plant-specific cleanliness program, -will allow for adequate core cooling.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.2-1 Item aa (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Refer to Ref. 1, Section 14.2.9.1.7, Expansion, Vibration, and Dynamic Effects Testing:

The test abstract describes preoperational testing for safety-related, high energy piping systems and components. The tests are required to verify that these systems and components are properly installed and supported consistent with the analytical basis for their design. These tests are designed to provide additional assurance that these safety-related systems have been adequately designed and constructed, and to verify that appropriate assumptions have been used in the analytical models for predicting system responses to anticipated transients and postulated design basis accidents.

Although the test abstract description included in the design control document (DCD) Tier 2 Material is adequate, there is no reference to any applicable Tier 1 Material (i.e., Inspections, Tests, Analyses and Acceptance Criteria (ITAAC)), nor does it appear that any ITAAC have been written to provide documentation of the test measurements for thermal expansion and for vibration amplitudes, and their comparison to allowable values.

Because of the safety significance of these systems, the corresponding tests of the as-built facility, and the comparison of test results to applicable acceptance criteria, should be included in the DCD Tier 1 Material to preclude potential departures from these requirements without appropriate NRC staff review. Please update the AP1000 DCD Tier 1 Material by providing appropriate ITAAC information for those systems, or portions thereof, described in Reference 1, Section 14.2.9.1.7, that will be subject to preoperational, hot functional testing prior to fuel load.

Reference 1: AP1000 DCD, APP-GW-GL-701 Rev.3, Tier 2 Material, dated January 2003.

Westinghouse Response:

The purpose of the expansion, vibration and dynamic effects testing is to verify that the safety-related, high energy piping and components are properly installed and supported such that expected movement due to thermal expansion during normal heatup and cooldown, and as a result of transients; thermal stratification and thermal cycling; as well as vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue to safety-related plant systems and equipment, as described in DCD section 3.9. In effect, this test is verifying that the ASME Class 1, 2 and 3 piping systems are designed consistent with the piping stress analyses that is performed to demonstrate conformance with the ASME code. The Tier 1 material includes suitable ITAAC to demonstrate that the as-built piping systems designated as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. This design commitment can be found in every Tier 1 system description that contains ASME Code piping. The ITAAC require that the ASME Code Section III design report

exists for the as-built piping. Therefore an additional ITAAC specifically for the expansion, vibration and dynamic effects testing is not added to the AP1000 Tier 1 material.

Please note that in accordance with accepted practice, the Tier 2 material does not reference Tier 1 material, except as discussed in DCD Section 14.3. In addition, the AP1000 Tier 1 material is consistent with the material that was identified as Tier 1 for the AP600, and is consistent with the approach that was taken for the other certified designs.

DCD Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.2-1 Item bb (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Refer to Ref. 1, Section 14.2.9.1.8, Control Rod Drive System

The test abstract describes preoperational testing for control rod drive mechanisms and associated motor-generator sets and control circuitry. These tests are required to provide assurance that construction and installation of essential, safety-related reactivity control components have been completed in accordance with the design.

Although the test abstract description included in the DCD Tier 2 Material is adequate, there is no reference to any applicable Tier 1 Material (i.e., ITAAC), nor does it appear that any ITAAC have been written to provide documentation of these tests including comparisons to specified acceptance criteria.

Because of the safety significance of these components, the corresponding tests of the as-built configuration and the comparison of test results to applicable acceptance criteria, should be included in the DCD Tier 1 Material to preclude potential departures from these requirements without appropriate NRC staff review. Please update the AP1000 DCD Tier 1 Material by providing appropriate ITAAC information for those components described in Reference 1, Section 14.2.9.1.8, that will be subject to preoperational, hot functional testing prior to fuel load.

Reference 1: AP1000 DCD, APP-GW-GL-701 Rev.3, Tier 2 Material, dated January 2003.

Westinghouse Response:

Testing of the control rod drive system is included in Tier 1 under 2.1.3 Reactor System. This section provides the Tier 1 Design Commitments and associated ITAAC for the control rod drive mechanisms.

Please note that in accordance with accepted practice, the Tier 2 material does not reference Tier 1 material, except as discussed in DCD Section 14.3. In addition, the AP1000 Tier 1 material is consistent with the material that was identified as Tier 1 for the AP600, and is consistent with the approach that was taken for the other certified designs.

DCD Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 15.2.7-1 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Through analyses performed for the AP600, it was established that the evolution of the LTC is independent of the initiating transient and the determining parameters are the decay heat, cooling water flow and steam-water flow resistance. At the initiation of the LTC, the core has been quenched, the accumulators and the CMTs have emptied, and the IRWST injection has stabilized. At this stage the objective is to demonstrate that the passive system is capable of removing the decay heat. Therefore, the limiting case has the highest decay heat, the lowest cooling water flow, and the highest resistance to steam-water mixture exiting the vessel. As in the AP600 the parameters of the limiting case occur in the DEDVI line break. DCD Tier 2 Section 15.6.5.4C.2 presents a DEDVI line break with an ADS-4 single failure. Initial containment pressure was derived from the WGOTHIC code, and the transient was carried to 9,000 seconds until after a quasi-steady state sump recirculation was established. The results are presented in DCD Tier 2 Figures 15.6.5.4C-1 through 15.6.5.4C-28. The results show that the fuel PCT is low, and the water circulation is adequate to provide core cooling.

With regard to the boron precipitation issue, the results presented in DCD transient analysis (1) did not quantify the amount of water exiting the vessel; (2) there was no clear indication of void distribution in the core; (3) did not characterize the water-steam mixture flow regime in the ADS-4; and (4) did not minimize the steam velocity through the ADS-4. At the staff's request, the applicant presented a more conservative case by assuming that all ADS-4 valves are open and the containment pressure is at a maximum. In addition, the applicant presented a qualification of the WCOBRA/TRAC model regarding ADS-4 water-steam flow (RAI responses to 440.091, Revision 1). The staff reviewed this information and (as stated above) found that there is adequate justification for the WCOBRA/TRAC ADS-4 flow model. The applicant demonstrated that the flow regime is the same as in AP600 (annular flow) which would entrain fluid particles to expel water from the vessel as required to avoid boron concentration in the vessel and/or precipitation. The amount of water to be removed from the core was quantified. In addition, literature was cited regarding flow regimes applicable to the conditions of the ADS-4 which reinforced the credibility of the results.

However, the applicant did not present a detailed enough case regarding void distribution in the core. Persistent voiding in the core could result into adiabatic heating of the fuel. This is Open Item 15.2.7-1.

Westinghouse Response Revision 1:

Summary

The original AP600/AP1000 WCOBRA/TRAC Long Term Cooling (LTC) model used in the DCD was based on a simplified noding. In particular, the core region was subdivided in [ ]^{a,c}.

Questions have been raised about the adequacy of such modeling, and in particular the axial core noding was judged to be insufficient to correctly model the core axial void fraction distribution.

As a result, the AP1000 LTC model was extended/modified as follows:

1. The Core was subdivided in [ ]^{a,c}.
2. The core region was subdivided axially in [ ]^{a,c} and is now consistent with nodalizations used to validate WCOBRA/TRAC against G1, G2 and FLECHT-SEASET tests.
3. The Upper Plenum explicitly models the CCFL region above the upper core plate and the nodalization is now equivalent to Westinghouse WCOBRA/TRAC LBLOCA model which was validated against full-scale UPTF tests.

Additional code validation was identified for the application of the revised WCOBRA/TRAC model to simulate the AP1000 LTC conditions. Selected G1 and G2 full-scale boil-off tests at pressure and power levels, which are prototypical of AP1000 conditions, were selected to validate the WCOBRA/TRAC core model. This validation included the determination, via sensitivity studies, of a corrective multiplier applied to the interfacial drag model such that the average core void fraction could be accurately predicted. Results from this validation are included in this response.

The validated model was then applied to simulate the LTC transient following a DEDVI break, which exhibits the most limiting relationship between core decay power (maximum) and available PXS liquid head (minimum).

The revised WCOBRA/TRAC analysis showed that adequate core cooling exists during the entire Long Term Cooling transient. The core inlet flow is more than sufficient to remove the decay heat and additional liquid is stored in the upper plenum and hot leg. No core temperature excursion is predicted to occur.

In addition a sensitivity study was performed where the interfacial drag coefficient was reduced by 20%. Results indicate that, under the AP1000 conditions, the core interfacial drag model has a negligible effect on the inner vessel mixture level. In both calculations (YDRAG=1.0 and YDRAG=0.8) mixture level is predicted in proximity of the hot leg centerline and the hot leg collapsed liquid level is almost identical in the two sensitivity cases.

These results are consistent with conclusions about the AP1000 system discussed in the response to Open Item 21.5-3. The analysis based on the simple AP1000 model showed that

the system draws more flow through the core than is needed to remove decay heat. Under those circumstances the mixture level is above the top of core and is virtually independent of the level swell model used within the core. In the AP1000 DEDVI event, during the long term cooling, the average core exit quality is indicated to be always less than 50%. This flow regime is quite different than a boil-off scenario such as in the G1 and G2 tests. In the boil-off mode the exit quality is approximately 1.0 and, once the two-phase mixture level drops below the top of the heated section, the rods are exposed to pure steam and can undergo an almost adiabatic heat-up. As a result, because of the sufficient liquid supply to the core, core heat-up does not occur during the AP1000 LTC phase following a LOCA event.

WCOBRA/TRAC Core Void Fraction Model Assessment Against G1 and G2 Low Pressure Boil-off Tests.

G1 test runs 28, 35, 38, 58 and 61 and G2 test runs 728, 729, 730, 732, 733, 734 were selected to validate the WCOBRA/TRAC core void fraction model used to perform the AP1000 long term cooling analysis. The following table shows the comparison between the test conditions and conditions expected in the AP1000 during the transient.

Test	Pressure (psia)		Power (kW/ft)		Core/Assembly Flow (in/sec)		Inlet Subcooling (F)	
AP1000	20	45	0.02	0.18	0.4	0.8	14	80
G1	[							]a,c
G2	[							]a,c

As discussed in the previous summary the AP1000 core is not to expected to be in a boil-off mode. Nevertheless, these experiments are useful to characterize the void fraction distribution and/or average void fraction within the core region when the mixture level is located above the top of the core.

G1 represents a prototypical [ ]a,b,c. G2 represents a [ ]a,b,c

For G1 the WCOBRA/TRAC Model includes the heated section, the lower plenum and the upper plenum and the downcomer region. The heated section is subdivided in [ ]a,b,c.

The boil-off test is initiated by setting the liquid level in the heated section and in the downcomer region to a given value. The power is turned on at the beginning of the test. The liquid in the lower plenum [

]a,b,c

The WCOBRA/TRAC model for G2 is very similar. In this case [

]a,b,c

At each given time, the location of the mixture level is defined by examining the rod temperature axial distribution. The rod surface temperature is close to saturation below the mixture level and suddenly increases significantly above the saturation temperature above the mixture level.

The average void fraction below the mixture level is related to a parameter called swell 'S' defined as follows:

a, b, c

[ ]

Figure 1 shows the measured swell compared to the swell predicted by the nominal WCOBRA/TRAC interfacial drag model. The swell (or average void fraction) tends to be over-predicted by the code.

a, b, c

[ ]

Figure 1

The G1 and G2 calculations were repeated by applying a multiplier (YDRAG=0.8) to the interfacial drag coefficient. Figure 2 shows the effect of a reduced interfacial drag. The predicted swell or void fraction is now in good agreement with the test data captured within ±20%. This multiplier was selected to be used in the revised WCOBRA/TRAC LTC analysis presented herein.

a, b, c

Figure 2

Results from the revised WCOBRA/TRAC model for the AP1000 Long Term Cooling phase following a DEDVI Break in PXS Room B.

The method used to analyze the AP1000 Long Term Cooling with WCOBRA/TRAC is described in the DCD and in the AP1000 code applicability document. The transient begins from the end of DEDVI analysis of NOTRUMP at 3000 seconds, and continues with boundary conditions provided by WGOTHIC (containment analysis) predictions.

Main results from the revised WCOBRA/TRAC LTC calculation are presented here and will be included in the revised DCD. The DCD includes a more detailed description of the transient. Here the discussion is limited to address the level swell issue and to derive some conclusions about the vessel liquid inventory which demonstrates that adequate cooling exists during the LTC.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **15.2.7-1 (Response Revision 1)**

Original RAI Number(s): **None**

Summary of Issue:

Through analyses performed for the AP600, it was established that the evolution of the LTC is independent of the initiating transient and the determining parameters are the decay heat, cooling water flow and steam-water flow resistance. At the initiation of the LTC, the core has been quenched, the accumulators and the CMTs have emptied, and the IRWST injection has stabilized. At this stage the objective is to demonstrate that the passive system is capable of removing the decay heat. Therefore, the limiting case has the highest decay heat, the lowest cooling water flow, and the highest resistance to steam-water mixture exiting the vessel. As in the AP600 the parameters of the limiting case occur in the DEDVI line break. DCD Tier 2 Section 15.6.5.4C.2 presents a DEDVI line break with an ADS-4 single failure. Initial containment pressure was derived from the WGOTHIC code, and the transient was carried to 9,000 seconds until after a quasi-steady state sump recirculation was established. The results are presented in DCD Tier 2 Figures 15.6.5.4C-1 through 15.6.5.4C-28. The results show that the fuel PCT is low, and the water circulation is adequate to provide core cooling.

With regard to the boron precipitation issue, the results presented in DCD transient analysis (1) did not quantify the amount of water exiting the vessel; (2) there was no clear indication of void distribution in the core; (3) did not characterize the water-steam mixture flow regime in the ADS-4; and (4) did not minimize the steam velocity through the ADS-4. At the staff's request, the applicant presented a more conservative case by assuming that all ADS-4 valves are open and the containment pressure is at a maximum. In addition, the applicant presented a qualification of the WCOBRA/TRAC model regarding ADS-4 water-steam flow (RAI responses to 440.091, Revision 1). The staff reviewed this information and (as stated above) found that there is adequate justification for the WCOBRA/TRAC ADS-4 flow model. The applicant demonstrated that the flow regime is the same as in AP600 (annular flow) which would entrain fluid particles to expel water from the vessel as required to avoid boron concentration in the vessel and/or precipitation. The amount of water to be removed from the core was quantified. In addition, literature was cited regarding flow regimes applicable to the conditions of the ADS-4 which reinforced the credibility of the results.

However, the applicant did not present a detailed enough case regarding void distribution in the core. Persistent voiding in the core could result into adiabatic heating of the fuel. This is Open Item 15.2.7-1.

Westinghouse Response Revision 1:

Summary

The original AP600/AP1000 WCOBRA/TRAC Long Term Cooling (LTC) model used in the DCD was based on a simplified noding. In particular, the core region was subdivided in [ ]^{a,c}.

Questions have been raised about the adequacy of such modeling, and in particular the axial core noding was judged to be insufficient to correctly model the core axial void fraction distribution.

As a result, the AP1000 LTC model was extended/modified as follows:

1. The Core was subdivided in [ ]^{a,c}.
2. The core region was subdivided axially in [ ]^{a,c} and is now consistent with nodalizations used to validate WCOBRA/TRAC against G1, G2 and FLECHT-SEASET tests.
3. The Upper Plenum explicitly models the CCFL region above the upper core plate and the nodalization is now equivalent to Westinghouse WCOBRA/TRAC LBLOCA model which was validated against full-scale UPTF tests.

Additional code validation was identified for the application of the revised WCOBRA/TRAC model to simulate the AP1000 LTC conditions. Selected G1 and G2 full-scale boil-off tests at pressure and power levels, which are prototypical of AP1000 conditions, were selected to validate the WCOBRA/TRAC core model. This validation included the determination, via sensitivity studies, of a corrective multiplier applied to the interfacial drag model such that the average core void fraction could be accurately predicted. Results from this validation are included in this response.

The validated model was then applied to simulate the LTC transient following a DEDVI break, which exhibits the most limiting relationship between core decay power (maximum) and available PXS liquid head (minimum).

The revised WCOBRA/TRAC analysis showed that adequate core cooling exists during the entire Long Term Cooling transient. The core inlet flow is more than sufficient to remove the decay heat and additional liquid is stored in the upper plenum and hot leg. No core temperature excursion is predicted to occur.

In addition a sensitivity study was performed where the interfacial drag coefficient was reduced by 20%. Results indicate that, under the AP1000 conditions, the core interfacial drag model has a negligible effect on the inner vessel mixture level. In both calculations (YDRAG=1.0 and YDRAG=0.8) mixture level is predicted in proximity of the hot leg centerline and the hot leg collapsed liquid level is almost identical in the two sensitivity cases.

These results are consistent with conclusions about the AP1000 system discussed in the response to Open Item 21.5-3. The analysis based on the simple AP1000 model showed that

the system draws more flow through the core than is needed to remove decay heat. Under those circumstances the mixture level is above the top of core and is virtually independent of the level swell model used within the core. In the AP1000 DEDVI event, during the long term cooling, the average core exit quality is indicated to be always less than 50%. This flow regime is quite different than a boil-off scenario such as in the G1 and G2 tests. In the boil-off mode the exit quality is approximately 1.0 and, once the two-phase mixture level drops below the top of the heated section, the rods are exposed to pure steam and can undergo an almost adiabatic heat-up. As a result, because of the sufficient liquid supply to the core, core heat-up does not occur during the AP1000 LTC phase following a LOCA event.

WCOBRA/TRAC Core Void Fraction Model Assessment Against G1 and G2 Low Pressure Boil-off Tests.

G1 test runs 28, 35, 38, 58 and 61 and G2 test runs 728, 729, 730, 732, 733, 734 were selected to validate the WCOBRA/TRAC core void fraction model used to perform the AP1000 long term cooling analysis. The following table shows the comparison between the test conditions and conditions expected in the AP1000 during the transient.

Test	Pressure (psia)		Power (kW/ft)		Core/Assembly Flow (in/sec)		Inlet Subcooling (F)	
AP1000	20	45	0.02	0.18	0.4	0.8	14	80
G1	[							]a,c
G2	[							]a,c

As discussed in the previous summary the AP1000 core is not to expected to be in a boil-off mode. Nevertheless, these experiments are useful to characterize the void fraction distribution and/or average void fraction within the core region when the mixture level is located above the top of the core.

G1 represents a prototypical [
]a,b,c. G2 represents a [
]a,b,c

For G1 the WCOBRA/TRAC Model includes the heated section, the lower plenum and the upper plenum and the downcomer region. The heated section is subdivided in [
]a,b,c.

The boil-off test is initiated by setting the liquid level in the heated section and in the downcomer region to a given value. The power is turned on at the beginning of the test. The liquid in the lower plenum [

]a,b,c

The WCOBRA/TRAC model for G2 is very similar. In this case [

]a,b,c

At each given time, the location of the mixture level is defined by examining the rod temperature axial distribution. The rod surface temperature is close to saturation below the mixture level and suddenly increases significantly above the saturation temperature above the mixture level.

The average void fraction below the mixture level is related to a parameter called swell 'S' defined as follows:

a, b, c

Figure 1 shows the measured swell compared to the swell predicted by the nominal WCOBRA/TRAC interfacial drag model. The swell (or average void fraction) tends to be over-predicted by the code.

a, b, c

Figure 1

The G1 and G2 calculations were repeated by applying a multiplier (YDRAG=0.8) to the interfacial drag coefficient. Figure 2 shows the effect of a reduced interfacial drag. The predicted swell or void fraction is now in good agreement with the test data captured within ±20%. This multiplier was selected to be used in the revised WCOBRA/TRAC LTC analysis presented herein.

a, b, c

Figure 2

Results from the revised WCOBRA/TRAC model for the AP1000 Long Term Cooling phase following a DEDVI Break in PXS Room B.

The method used to analyze the AP1000 Long Term Cooling with WCOBRA/TRAC is described in the DCD and in the AP1000 code applicability document. The transient begins from the end of DEDVI analysis of NOTRUMP at 3000 seconds, and continues with boundary conditions provided by WGOTHIC (containment analysis) predictions.

Main results from the revised WCOBRA/TRAC LTC calculation are presented here and will be included in the revised DCD. The DCD includes a more detailed description of the transient. Here the discussion is limited to address the level swell issue and to derive some conclusions about the vessel liquid inventory which demonstrates that adequate cooling exists during the LTC.

The time scale of the plots herein is adjusted to reflect DEDVI break transient time. Figure 3 shows the upper plenum pressure. The pressure decreases from its initial value to reach a quasi steady state value of 28 psia at about 7000 seconds.

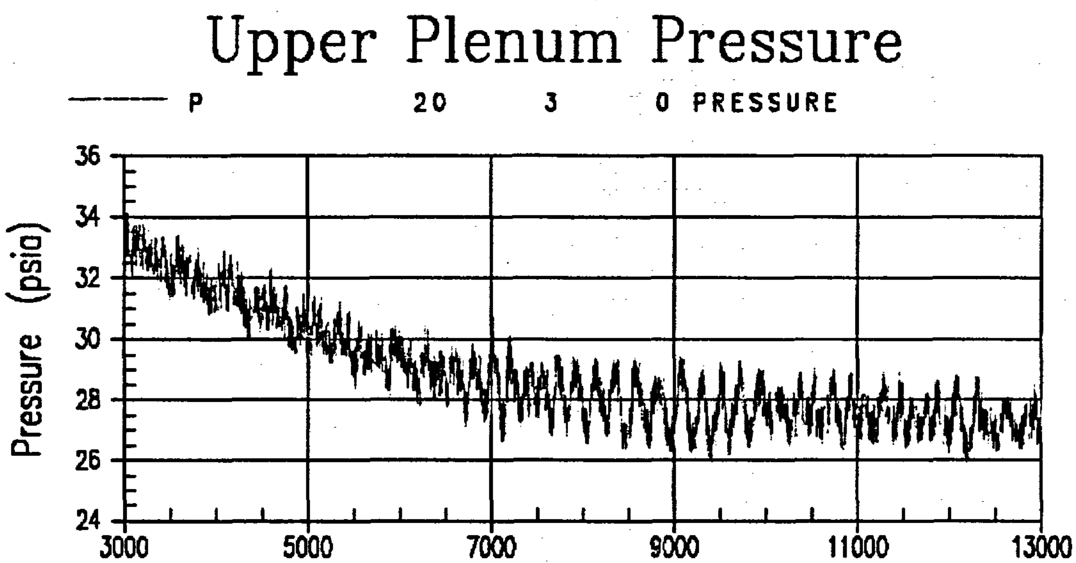

Figure 3

The next figures 4 and 5 show the ADS4 integrated flows and the integrated flows from the DVI nozzles:

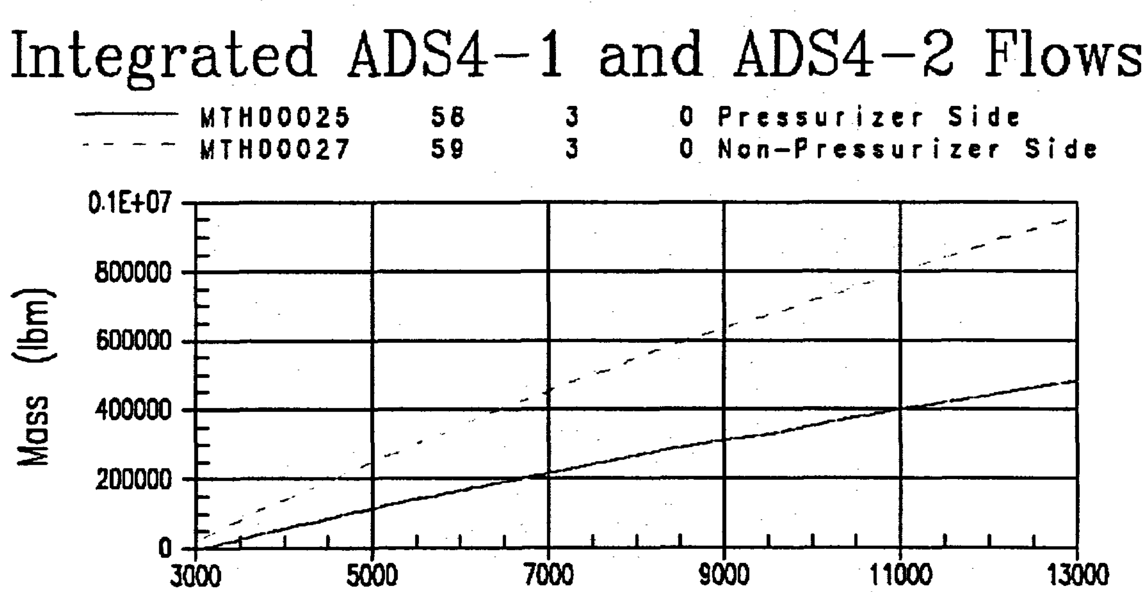

Figure 4

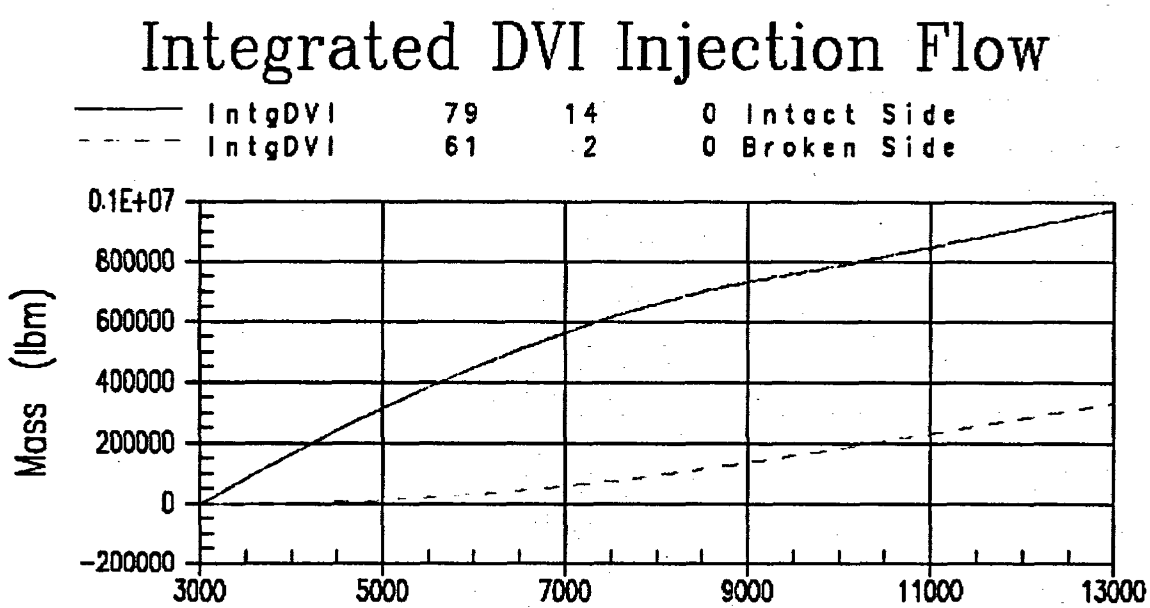

Figure 5

The inner vessel collapsed liquid level as well as the core region only collapsed liquid level are shown in Figures 6 and 7:

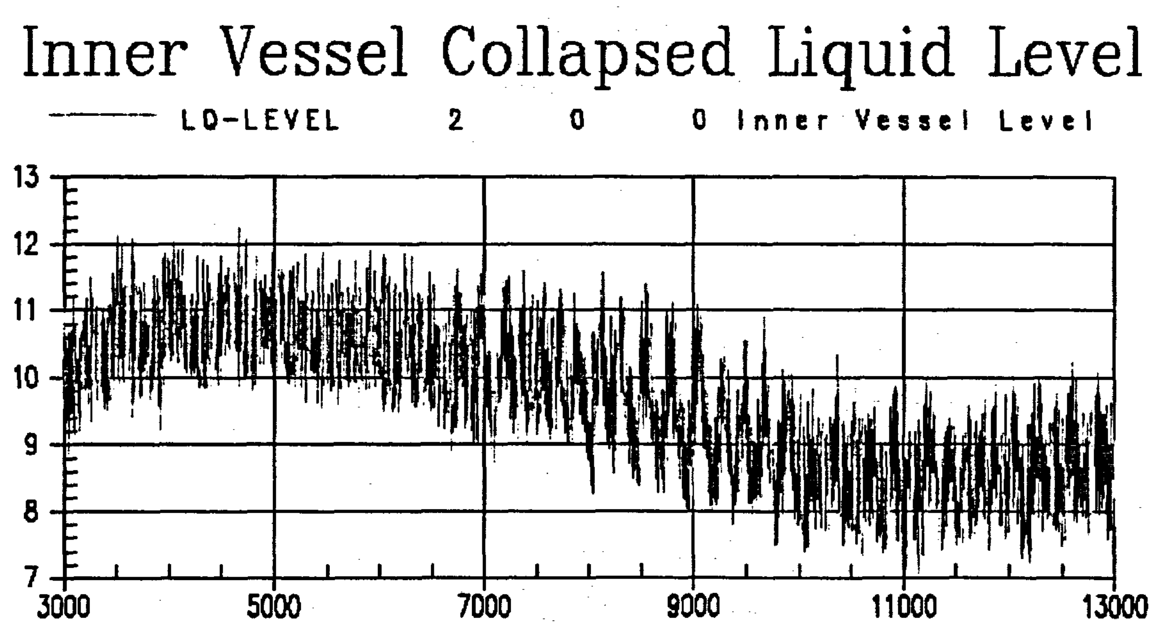

Figure 6

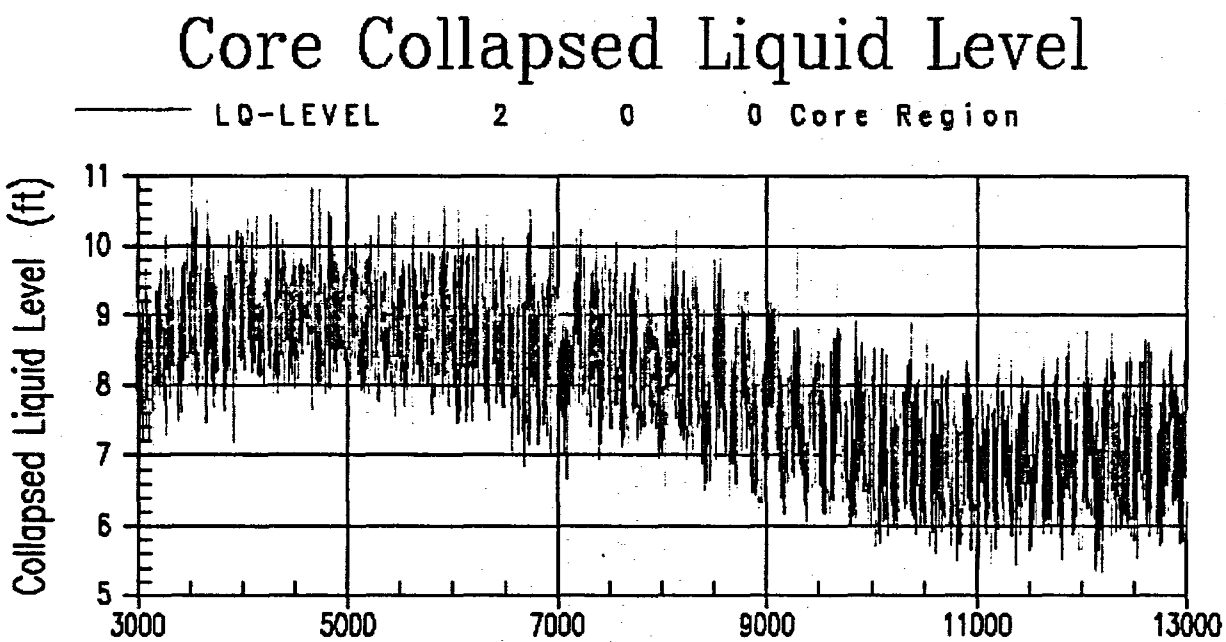

Figure 7

Figure 8 shows that the mixture level is located in proximity of the hot leg centerline.

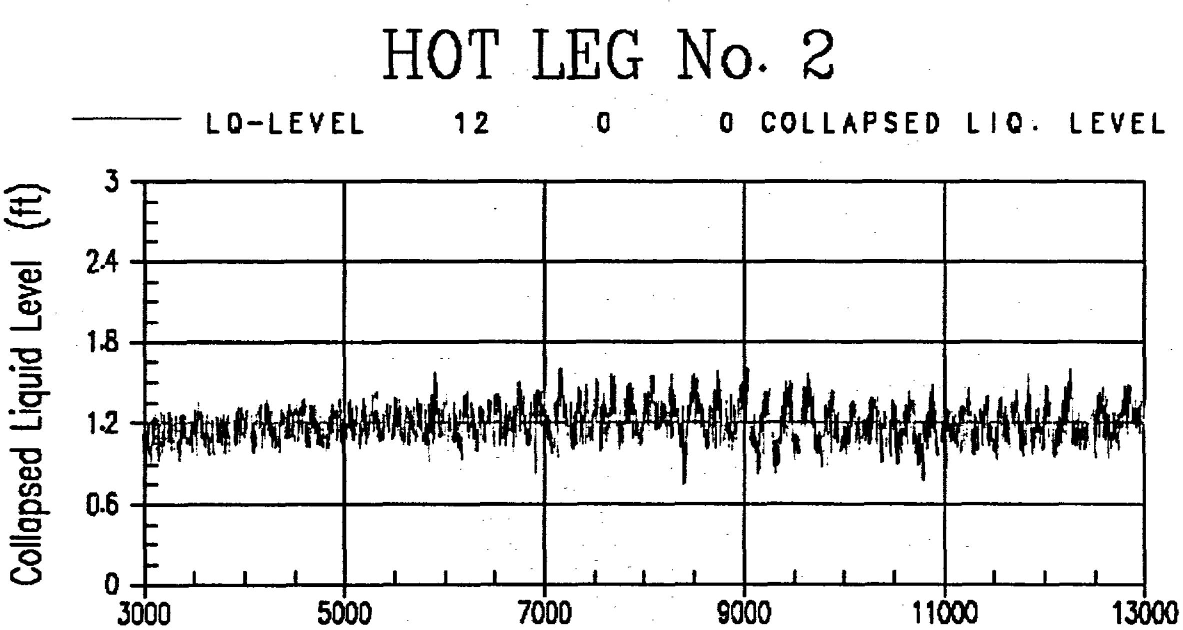

Figure 8

It is worth to note that the LTC case was analyzed with both nominal interfacial drag model and with 20% reduced interfacial drag model and it was observed that the hot leg levels from these calculations were nearly identical as shown in Figure 9.

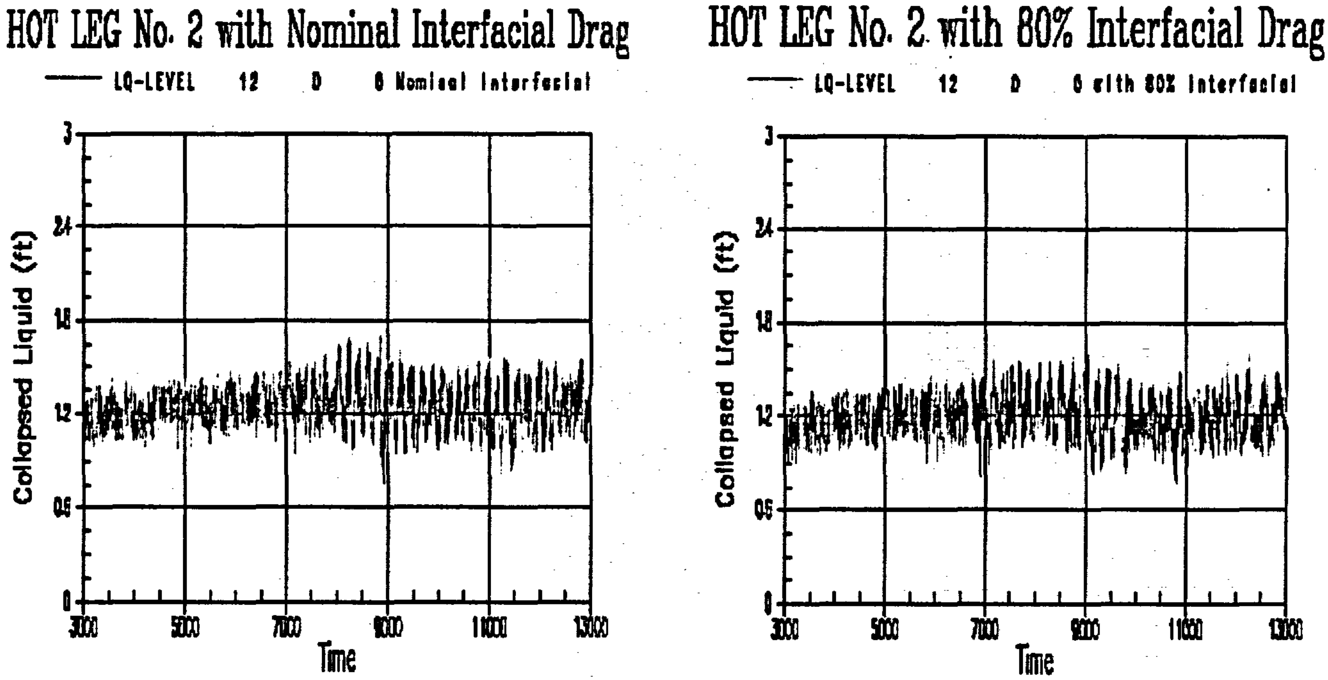

Figure 9

This result is an indication that once the mixture level is located above the top of the core and well into the upper plenum, the interfacial drag model or core swell model has a very small effect on the overall system behavior.

The liquid supply (core inlet liquid flow) is always sufficient to remove the decay heat. Additional liquid is stored in the upper plenum and discharged by the ADS-4. Figure 10 shows that the ADS-4 average exit quality float around 50% during the LTC transient.

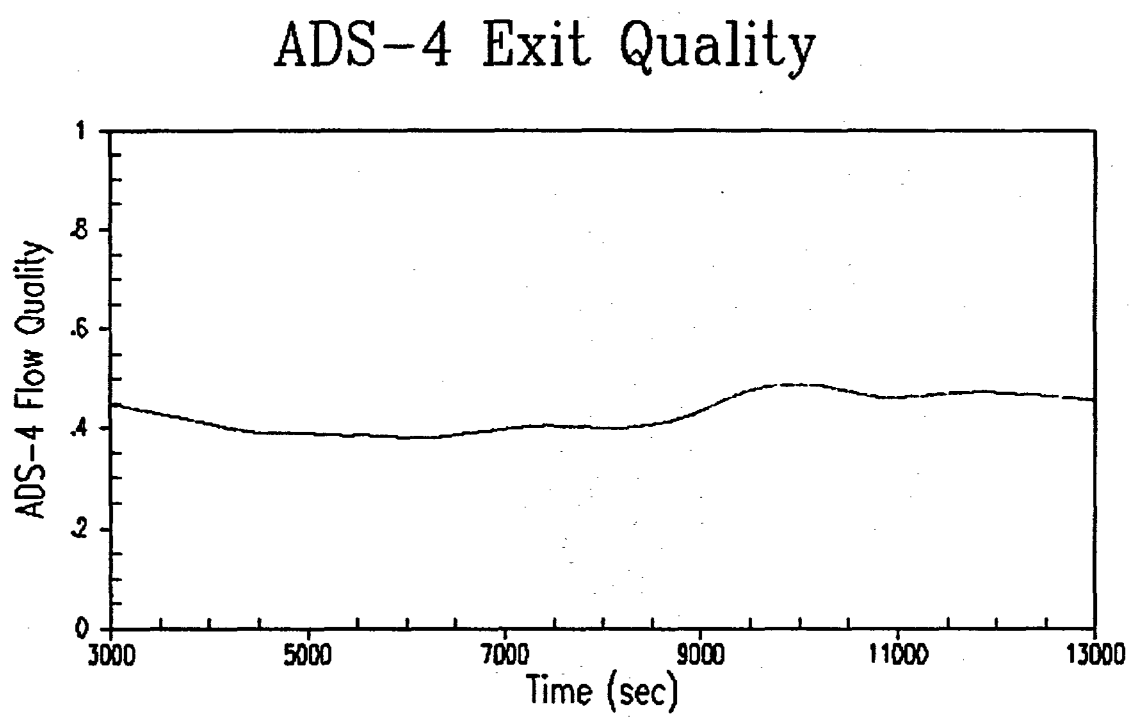

Figure 10

The predicted void fraction at the top of the core hot assembly is approximately 0.8 during the transient (Figure 11) which is another indication that sufficient liquid is provided at the top of the core preventing core heat-up from occurring.

Figure 12 shows that the clad temperature in the top region of the core is always close to the saturation temperature and no heat-up excursion is predicted to occur.

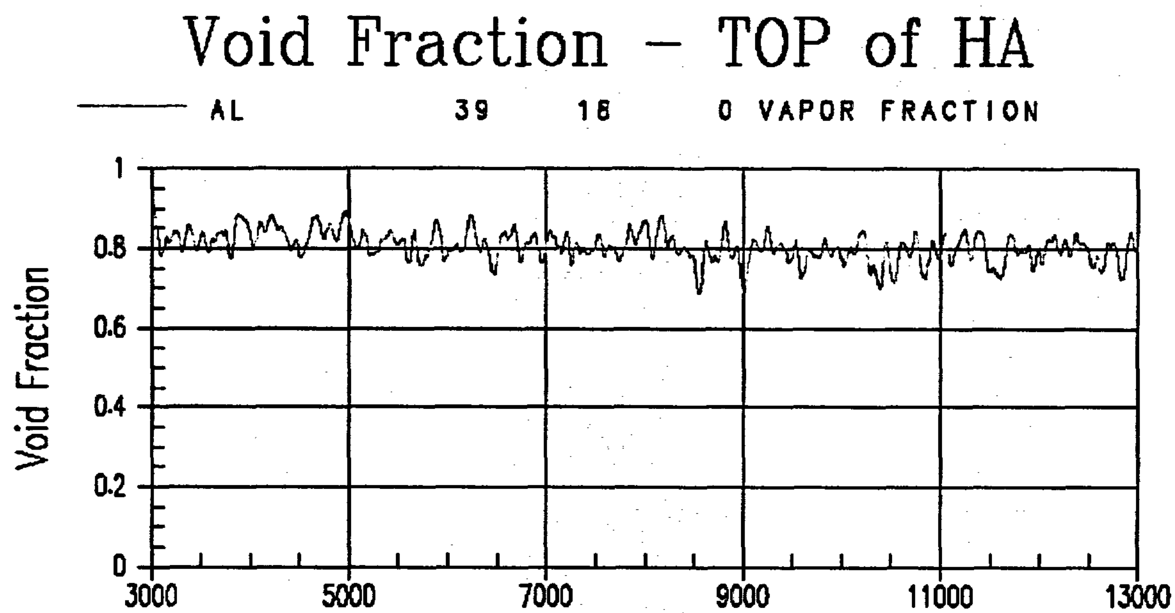

Figure 11

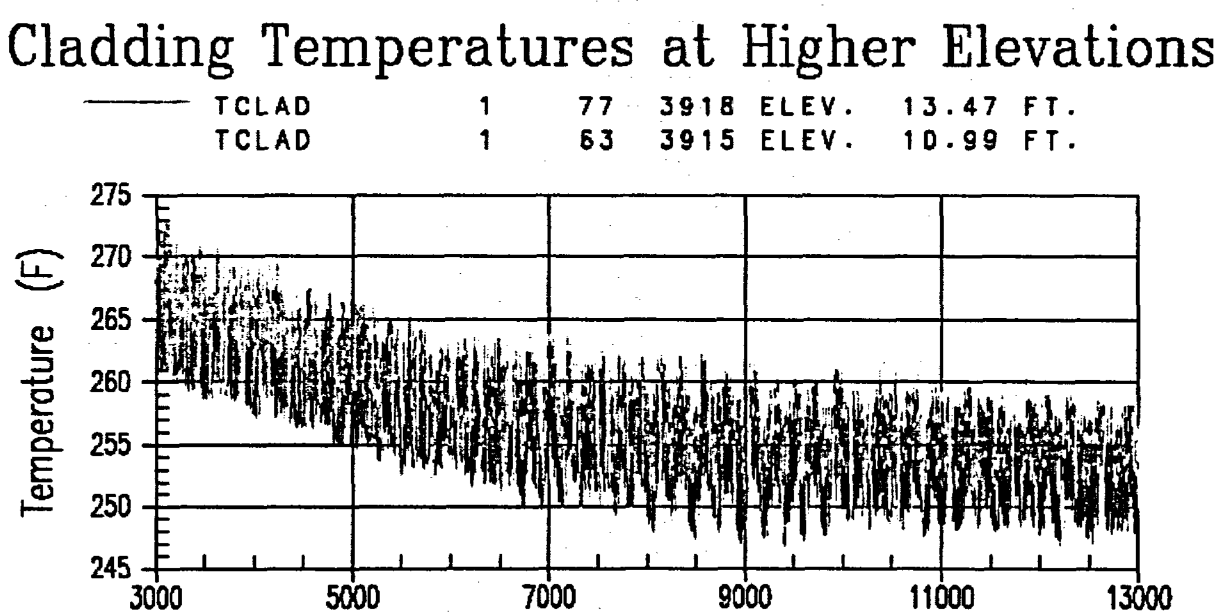

Figure 12

Additional Considerations

Further investigations were made to establish what flow regime should be expected in the top region of the core to further support that under the conditions expected during the LTC, adequate core cooling is provided to prevent core heat-up from occurring.

The expected flow regime at the top of the core is a churn or pulsated annular flow. The steam velocity is so low that entrainment of droplets is not expected to occur. Based on Ishii and Grolmes (1975) inception criteria for droplet entrainment in two-phase concurrent film (roll wave and liquid jet instabilities) the critical superficial velocity for droplet entrainment was estimated to be 77 ft/s (P=40 psia). Yonomoto et. al. (1987) (JAERI) established a criterion for entrainment onset based on reflood tests in rod bundle prototypical geometries and conditions. Based on Yonomoto model the onset is at about 20 ft/s at the same conditions. During the LTC, the vapor superficial velocity at the core exit is expected to be lower than 16 ft/s.

The possibility that the CHF could be exceeded, below the two-phase mixture level was also investigated. Schosse et. al. (1997) presented a review of CHF correlations applicable to low upward flows near atmospheric pressure. It was found that the AP1000 typical heat flux (the average heat flux is about 1.0 Btu/ft^2-s at 3000 sec.) is significantly less than the critical heat flux which can be predicted with their model.

References

1. Ishii, M. and Grolmes, M. A. (1975), Inception Criteria for Droplet Entrainment in Two-Phase Concurrent Film Flow, A.I.Ch.E. Jl 21, 308.

2. Yonomoto, T. et. al. (1987). Liquid Entrainment for Liquid Entrainment in Reflooding phase of LOCA. J. of Nuclear Science and Technology. Vol. 24 [10].

3. Schoesse, T. et. al. (1997), Critical Heat Flux in a Vertical Annulus under Low Upward Flow and near Atmospheric Pressure. J. of Nuclear Science and Technology. Vol. 34 [6].

Design Control Document (DCD) Revision:

Revision 1 of this response provides a complete markup of DCD subsection 15.6.5.4C, including an additional LTC analysis case (wall-to-wall floodup) and the revisions associated with the response to boron precipitation during the LTC phase. The revised Figures 15.6.5.4C-1 through 15.6.5.4C-28 are included.

15.6.5.4C Post-LOCA Long-Term Cooling

15.6.5.4C.1 Long-Term Cooling Analysis Methodology

The AP1000 safety-related systems are designed to provide adequate cooling of the reactor indefinitely. Initially, this is achieved by discharging water from the IRWST into the vessel. When the low-3 level setpoint is reached in the IRWST, the containment recirculation subsystem isolation valves open and water from the containment reactor coolant system (RCS) compartment can flow into the vessel through the PXS piping. The water in containment rises in temperature toward the saturation temperature. Long-term heat removal from the reactor and containment is by heat transfer through the containment shell to atmosphere.

The purpose of the long-term cooling analysis is to demonstrate that the passive systems provide adequate emergency core cooling system performance during the IRWST injection/containment recirculation time scale. The long-term cooling analysis is performed using the WCOBRA/TRAC computer code to verify that the passive injection system is providing sufficient flow to the reactor vessel to cool the core and to preclude boron precipitation.

The AP1000 long-term cooling analysis is supported by the series of tests at the Oregon State University AP600 APEX Test Facility. This test facility is designed to represent the AP600 reactor safety-related systems and nonsafety-related systems at quarter-scale during long-term cooling. The data obtained during testing at this facility has been shown to apply to the AP1000 (Reference 25). These tests were modeled using WCOBRA/TRAC with an equivalent noding scheme to that used for AP~~1000~~600 (Reference 17~~23~~) in order to validate the code for long-term cooling analysis.

Reference ~~17~~24 provides details of the AP~~6~~1000 WCOBRA/TRAC modeling. The coarse reactor vessel modeling used for AP600 has been replaced with a detailed noding like ~~is much coarser than~~ that applied in the large-break LOCA analyses described in subsection 15.6.5.4A. ~~to permit faster computation and because less detail is required for the slowly changing parameters involved for the long-term transient.~~ The ~~equivalent~~ reactor vessel noding ~~is~~ used in the AP1000 long-term cooling analyses in core and upper plenum regions is equivalent to that used in full-scale test simulations (see Reference 24).

A DEDVI line break is analyzed because it is the most limiting long-term cooling case in the relationship between decay power and available liquid driving head. Because the IRWST spills ~~the transfer~~ directly onto the containment floor in a DEDVI break, this event has the highest core decay power ~~to sump injection~~ when the transfer to sump injection is initiated. In postulated DEDVI break cases, ~~before~~ the compartment water level exceeds the elevation at which the DVI line enters the reactor vessel, so water can flow from the containment into the reactor vessel through the broken DVI line; this in-flow of water through the broken DVI line assists in the heat removal from the core. The steam produced by boiling in the core vents to the containment through the ADS valves and condenses on the inner surface of the steel containment vessel. The condensate is collected and drains to the IRWST to become available for injection into the reactor coolant system. The WCOBRA/TRAC analysis presented analyzes the DEDVI small-break LOCA event from a time (3000 seconds) at which IRWST injection is fully established to ~~to~~ beyond the time of containment recirculation. During this time, the head of water to drive the flow into the vessel for IRWST injection decreases from the initial level to its lowest value at the containment recirculation switchover time. ~~T~~PXS Room -B is the location of the break in the DVI line ~~is adjacent to~~

~~the vessel nozzle~~. **At this break location**~~, the in-flow through the broken DVI line is minimized as the IRWST drains~~ **liquid level in containment at the time of recirculation is a minimum.**

A continuous analysis of the post-LOCA long term cooling is provided from the time of stable IRWST injection through the time of sump recirculation for the DEDVI break. Maximum design resistances are applied in WCOBRA/TRAC for both the ADS Stage 4 flow paths and the IRWST injection and containment recirculation flow paths.

The break modeled is a double-ended guillotine rupture of one of the direct vessel injection lines. The long-term cooling phase begins after the simultaneous opening of the isolation valves in the IRWST DVI lines and the opening of ADS Stage 4 squib valves, when flow injection from the IRWST has been fully established. Initial conditions are taken from the NOTRUMP DEDVI case at 25 psia containment pressure reported in subsection 15.6.5.4B.

15.6.5.4C.2 DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case

This subsection presents the results of a DEDVI line break analysis during IRWST injection phase continuing into sump recirculation. Initial conditions at the start of the case are prescribed based on the NOTRUMP DEDVI break results to allow a calculation to begin shortly after IRWST injection begins in the small break long-term cooling transient. The WCOBRA/TRAC calculation is then allowed to proceed until a quasi-steady-state is achieved. At this time, the predicted results are independent of the assumed initial conditions. This calculation uses boundary conditions taken from a WGOTHIC analysis of this event. During the calculation, which is carried out for 10,000 seconds until a quasi-steady-state sump recirculation condition has been established, the IRWST water level is decreased continuously until the sump recirculation setpoint is reached.

In the analysis, one of the two ADS Stage 4 valves in the PRHR loop is assumed to have failed. The initial reactor coolant system liquid inventory and temperatures are determined from the NOTRUMP calculation. ~~This equates to a full lower plenum and downcomer, a core collapsed liquid level of 9.5 feet (relative to the bottom of the heated length), and a collapsed level of 2.2 feet in the upper plenum.~~ The core makeup tanks do not contribute to the DVI injection **during this phase of the transient.** Steam generator secondary side conditions are taken from the NOTRUMP calculation (at the beginning of long-term cooling). The reactor coolant pumps are tripped and not rotating.

The levels and temperatures of the liquid in the containment sump and the containment pressure are based on ~~a~~ WGOTHIC calculations of the conservative minimum pressure during this long-term cooling transient, using the methodology described in Reference 17. Small changes in the RCS compartment level do not have a major effect on the predicted core collapsed liquid level or on the predicted flow rate through the core. ~~Sensitivity studies for this break scenario, which ranged level from 110 feet to 109.4 feet, predicted adequate core cooling during this time window.~~ The minimum compartment floodup level for this break scenario is 1097.48 feet or greater.

In this transient, the IRWST provides a hydraulic head sufficient to drive water into the downcomer through the intact DVI nozzle. Also, water flows into the downcomer from the RCS loop compartment through the broken DVI line once the liquid level is adequate to support flow. The water flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core and liquid flow

out of the reactor coolant system via the ADS Stage 4 valves. There is little flow out of ADS Stages 1, 2, and 3 even when the IRWST liquid level falls below the sparger elevation, so they are not modeled in this calculation. The venting provided by the ADS-4 paths enables the liquid flow through the core to maintain core cooling.

Approximately 500 seconds of WCOBRA/TRAC calculation are required to establish the quasi-steady-state condition associated with IRWST injection at the start of long-term cooling and so are ignored in the following discussion. The hot leg levels are such that during the IRWST injection phase the quality of the ADS Stage 4 mass flows varies as water is carried out of the hot legs. This periodically increases the pressure drop across the ADS Stage 4 valves and the upper plenum pressure. The higher pressure in the upper plenum reduces the injection flow. This cycle of pressure variations due to changing void fractions in the flow through ADS Stage 4 is consistent with test observations and is expected to recur often during long-term cooling.

The head of water in the IRWST causes a flow of subcooled water into the downcomer at an approximate rate of 1780 lbm/~~sec~~ through the intact DVI nozzle at the start of long-term cooling. The downcomer level at the end of the code initiation (the start of long-term cooling) is about ~~19~~18.5 feet (Figure 15.6.5.4C-1). Note that the time scale of this and other figures in subsection 15.6.5.4C.2 is offset by 2500 seconds; that is, a time of 500 seconds on the Figure 15.6.5.4C-1 axis equals 3000 seconds transient time for the DEDVI break. All of the injection water flows down the downcomer and up through the core. The accumulators have been fully discharged before the start of the time window and do not contribute to the DVI flow.

Boiling in the core produces steam and a two-phase mixture, which flows into the upper plenum. The core is 14 feet high, and the core average collapsed liquid level (Figure 15.6.5.4C-2) is ~~about 7.7 feet at~~ shown from the start of long-term cooling. The boiling process causes a variable rate of steam production and resulting pressure changes, which in turn causes oscillations in the liquid flow rate at the bottom of the core and also variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC noding, the core is divided ~~into two axial levels, each of which is 7 feet high~~ both axially and radially as described in Reference 24. The void fractions in the top two ~~levels~~ cells of the hot assembly are shown as Figures 15.6.5.4C-3 and -4. ~~The core void fraction is low for the bottom cell and has a mean void fraction less than 0... for the entire long-term cooling transient.~~ The average void fraction of these upper core cells is about ~~exceeds~~ 0.8 during ~~at the start of~~ long-term cooling, during IRWST injection, and ~~decreases as the decay heat decreases, and levels out in the~~ into the containment recirculation period. ~~that begins at 6700 seconds on the figures.~~ There is a continuous flow of two-phase fluid into the hot legs, and mainly vapor flow toward the ADS Stage 4 valve occurs at the top of the pipe. The collapsed liquid level in the hot leg varies between ~~1~~0.9 feet to 1.~~7~~5 feet (Figure 15.6.5.4C-5). The hot legs on average are more than 50-percent full. Vapor and liquid flows at the top of the core are shown in Figures 15.6.5.4C-6 and 15.6.5.4C-7, the upper plenum collapsed liquid level in Figure 15.6.5.4C-8. Figures 15.6.5.4C-9 and 15.6.5.4C-10 are ADS stage 4 mass flowrates.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-11. The upper plenum pressure fluctuation that occurs is due to the ADS Stage 4 water discharge. The ~~hot rod~~ PCT at the top of the hot assembly rod follows saturation temperature (Figure 15.6.5.4C-12), which demonstrates that the calculated ~~core~~ core collapsed liquid level is adequate to ~~r~~provide enough liquid at the top of the core that no uncovery and no cladding temperature excursion occurs. A small pressure drop is

calculated across the reactor vessel, and injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-13 and -14. Figure 15.6.5.4C-1~~3~~4 shows the flow is outward through the broken DVI line at the start of the long-term cooling period, and it increases to a maximum average value of about 529~~0~~ lbm/s after the compartment water level has increased above the nozzle elevation to permit liquid injection into the reactor vessel. In contrast, the intact DVI line~~-B~~ flow falls from 17~~8~~0 lbm/s with a full IRWST to about ~~80~~65 lbm/s flow from the containment at the end of the calculation. The recirculation core liquid throughput is more than adequate to preclude any boron buildup on the fuel.

15.6.5.4C.3 DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation

This subsection presents a DEDVI line break analysis with wall-to-wall flooding due to leakage between compartments, using the window mode methodology. All containment free volume beneath the level of the liquid is assumed filled in this calculation to generate the minimum water level condition during containment recirculation. The time identified for this calculation is ~~28.5~~14 days into the event, and the core power is calculated accordingly. The initial conditions at the start of the window are consistent with the analysis described in subsection 15.6.5.4C.2. Containment recirculation is simulated during the time window. The calculation is ~~then~~ carried out over ~~3000 seconds, which is~~ a time period long enough to establish a quasi-steady-state solution; after 1000 seconds of problem time, the flow dynamics are quasi-steady-state and the predicted results are independent of the assumed initial conditions. The liquid level is simulated constant at 28.~~4~~29 feet~~"~~ above the bottom inside surface of the reactor vessel (refer to Figure 15.0.3-2 for AP1000 reference plant elevations) during the time window, and~~while~~ the liquid temperature in containment is 205°F~~set at the saturation condition at~~. ~~t~~The identified containment pressure is 19.0~~of 32.7~~ psia. The single failure of an ADS Stage 4 flow path is assumed as in the subsection 15.6.5.4C.2 case.

Focusing on the 1000- to ~~3000~~4000–second time interval of this case, the containment liquid provides a hydraulic head sufficient to drive water into the downcomer through the DVI nozzles. The water introduced into the downcomer flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core entrains liquid and flows out of the reactor coolant system via the ADS Stage 4 valves. The DVI flow and the venting provided by the ADS paths provide a liquid flow through the core that enables the core to remain cool.

The downcomer collapsed liquid level (Figure 15.6.5.4C-15) is almost constant during the transient at about 2~~5~~4 feet, just below the lower lip of the ~~cold leg~~DVI line nozzles. Pressure spikes produced by boiling in the core can cause the mass flow of the DVI flow rates shown in Figures 15.6.5.4C-27 and -28 into the vessel to stop momentarily, but the injection flow is quickly reestablished.

Boiling in the core produces steam and a two-phase mixture, which flows out of the core into the upper plenum. The core is 14 feet high, and the core collapsed liquid level (Figure 15.6.5.4C-16) maintains a mean level close to the top of the core. The boiling process causes pressure variations, which in turn, cause~~s~~ variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC ~~noding~~analysis, the core is ~~divided into two axial levels, each 7 feet long~~noddalized as described in Reference 24. The void fraction in the top ~~level~~cell is shown in Figure 15.6.5.4C-~~18~~17 for the core hot assembly, and ~~, while~~ Figure 15.6.5.4C-~~17~~18 shows the ~~small~~ void fraction that exists ~~at the bottom level~~one cell further down in the hot assembly. The PCT does not rise appreciably above the saturation temperature (Figure 15.6.5.4C.3-26) at the top of the hot rod. The flow through the core and out of the reactor coolant system is more than sufficient to provide adequate flushing

to preclude concentration of the boric acid solution. Liquid collects above the upper core plate in the upper plenum, where the average collapsed liquid level is about ~~4.6~~3.6 feet (Figure 15.6.5.4C-22). There is no significant flow through the cold legs into either the broken or the intact loops, and there is no significant quantity of liquid residing in any of the cold legs.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-25. The upper plenum pressurization, which occurs periodically, is due to the ADS Stage 4 water discharge. The collapsed liquid level in the hot leg of the pressurizer loop varies between ~~0.0~~1.0 feet to ~~2.0~~2.2 feet, as shown in Figure 15.6.5.4C-19. Injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-27 and -28.

15.6.5.4C.4 Long-Term Core Boron Concentration

For the AP1000, water carryover out the ADS Stage 4 lines limits the potential core boron concentration buildup following a cold leg LOCA. The higher the ADS Stage 4 vent quality, the higher the core boron concentration buildup. Analyses have been performed to bound the maximum core boron concentration buildup.

These analyses demonstrate that highest ADS Stage 4 vent qualities result from the following:

- Highest decay heat levels
- Lowest PXS injection-/ADS 4 vent flows, including high line resistances and low containment water levels

The LTC analysis discussed in subsection 16.6.5.4C.2 is consistent with these assumptions. The ADS Stage 4 vent quality resulting from this analysis is less than 40 percent at the beginning of IRWST injection and reaches a maximum of less than 50 percent around the initiation of recirculation. It decreases after this peak, dropping to a value less than 8 percent at 14 days.

With high decay heat values, the ADS Stage 4 vent flows and velocities are high. These high vent velocities result in flow regimes that are annular out through at least 14 days and slug/churn after that time. Such flow regimes can move water up and out the ADS Stage 4 lines. These flow regimes are based on the Taitlel-Dukler vertical flow regime map. Lower decay heat levels can also be postulated later in time or just after a refueling outage. Significantly lower decay heat levels result in lower ADS Stage 4 vent qualities. They also result in ADS Stage 4 vent flows/ velocities that are much lower. With low ADS 4 vent flow velocities, the AP1000 plant will operate as a manometer. The small amount of steam generated in the core is sufficient to reduce the density of the steam/water mixture in the ADS Stage 4 line and allow the injection head to push the steam/water mix out the ADS Stage 4 line. The limiting condition for core boron buildup is with high decay heat that leads to the highest ADS Stage 4 vent qualities.

With the maximum ADS Stage 4 vent qualities, the maximum core boron concentration peaks at a value less than 7400 ppm at the time of recirculation initiation. After this time, the core boron concentration decreases as the ADS Stage 4 vent quality decreases, reaching 5000 ppm about 6 hours after the accident. The core boron solubility temperature reaches a maximum of 58°F (at 7400 ppm) and quickly drops to 40°F (at 5000 ppm). With these low core boron solubility temperatures, there is no concern with cold PXS injection water causing boron precipitation in the

core. With the IRWST located inside containment, its water temperature is normally expected to be above these solubility temperatures. The minimum core inlet temperature is greater than 120°F considering the minimum IRWST temperature permitted by the Technical Specifications (50°F) and the heatup of the injection by steam condensation and pickup of sensible heat from the reactor vessel, core barrel, and lower support plate.

The boron concentration water in the containment is initially about 2980 ppm. As the core boron concentration increases, the containment concentration decreases slightly. The minimum boron concentration in containment is greater than 2950 ppm. The solubility temperature of the containment water at its maximum boron concentration is 32°F.

15.6.5.4C.54 Conclusions

Calculations of AP1000 long-term cooling performance have been performed using the WCOBRA/TRAC model ~~approved for AP600~~developed for AP1000 and described in Reference 24. The DEDVI case was chosen because it reaches sump recirculation at the earliest time (and highest decay heat). A window mode case at the minimum containment water level ~~6-month~~postulated to occur 2 weeks into long-term cooling was also performed.

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

15.6.6 References

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

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

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

4. Dederer, S. I., Hochreiter, L. E., Schwartz, W. R., Stucker, D. L., Tsai, C.K., and Young, M. Y., "Westinghouse Large-Break Best Estimate Methodology, Volume 1 Model Description and Validation, Volume 2, Revision 2, Application to Two-Loop PWRs Equipped with Upper Plenum Injection," WCAP-10924-P-A (Proprietary), December 1988.

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

6. DELETED

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

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

9. DELETED

10. Bajorek, S. M., Young, M. Y., Dederer, S. I., Nissley, M. E., Takeuchi, K., and Ohkawa, K., "Code Qualification Document for Best Estimate Analysis," Volumes 1 through 5, WCAP-12945-P, Revision 2 (Proprietary), March 1998.

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

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

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

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

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

16. Friedland, A. J., Ray S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11398-A (Nonproprietary), April 1989.

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

18. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908 (Proprietary) and WCAP-7337 (Nonproprietary), June 1975.

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

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

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

22. "NOTRUMP Final Validation Report for AP600," WCAP-14807, Revision 5, (P), August 1998.

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

24. "AP1000 Code Applicability Report," WCAP-15644 (Proprietary) and WCAP-15707 (Nonproprietary), May 2001.

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

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

27. "WCOBRA/TRAC AP1000 ADS-4/IRWST Phase Modeling," WCAP-15833 (Proprietary) and WCAP-15834 (Nonproprietary) April 2002.

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

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

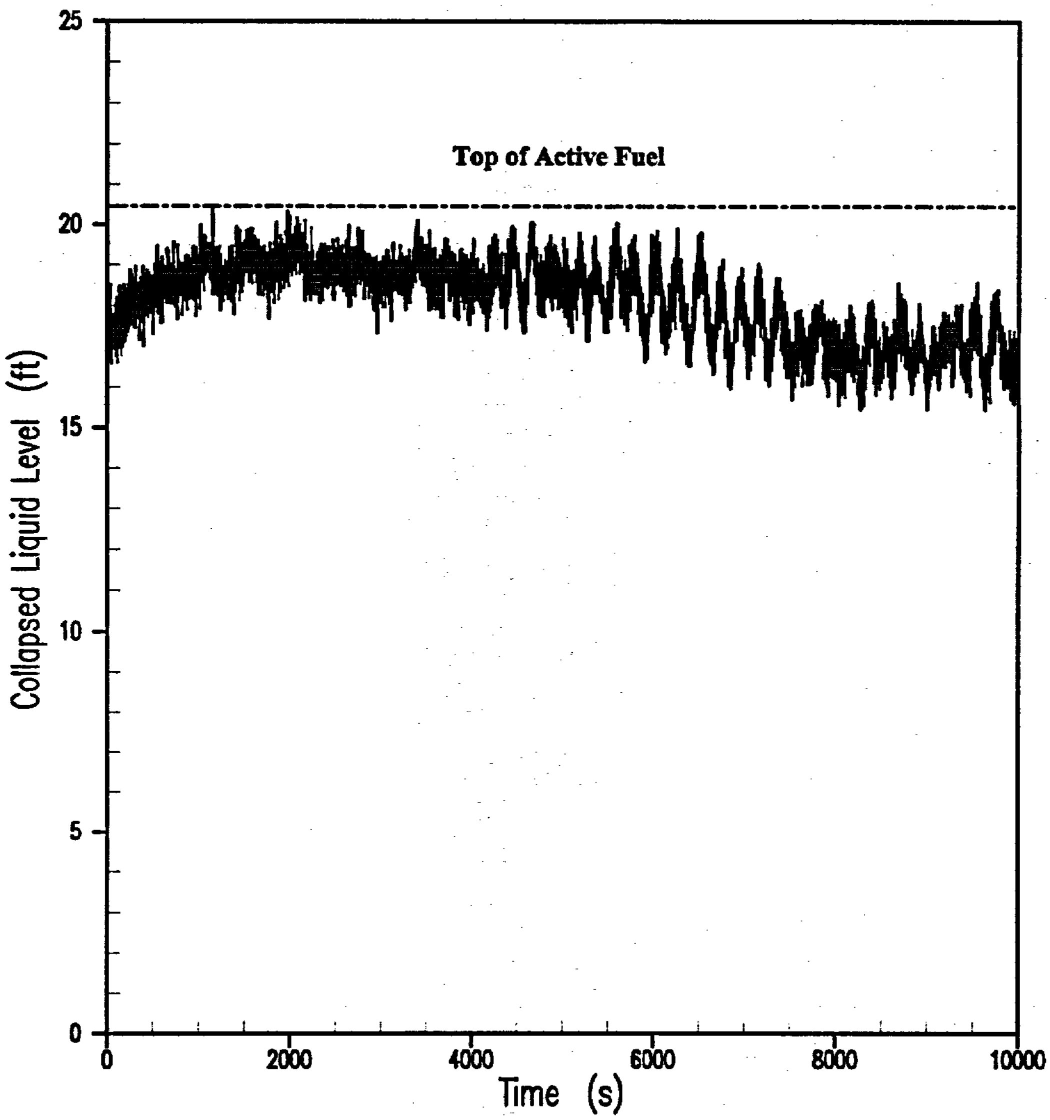

Figure 15.6.5.4C-1

Collapsed Level of Liquid in the Downcomer (DEDVI Case)

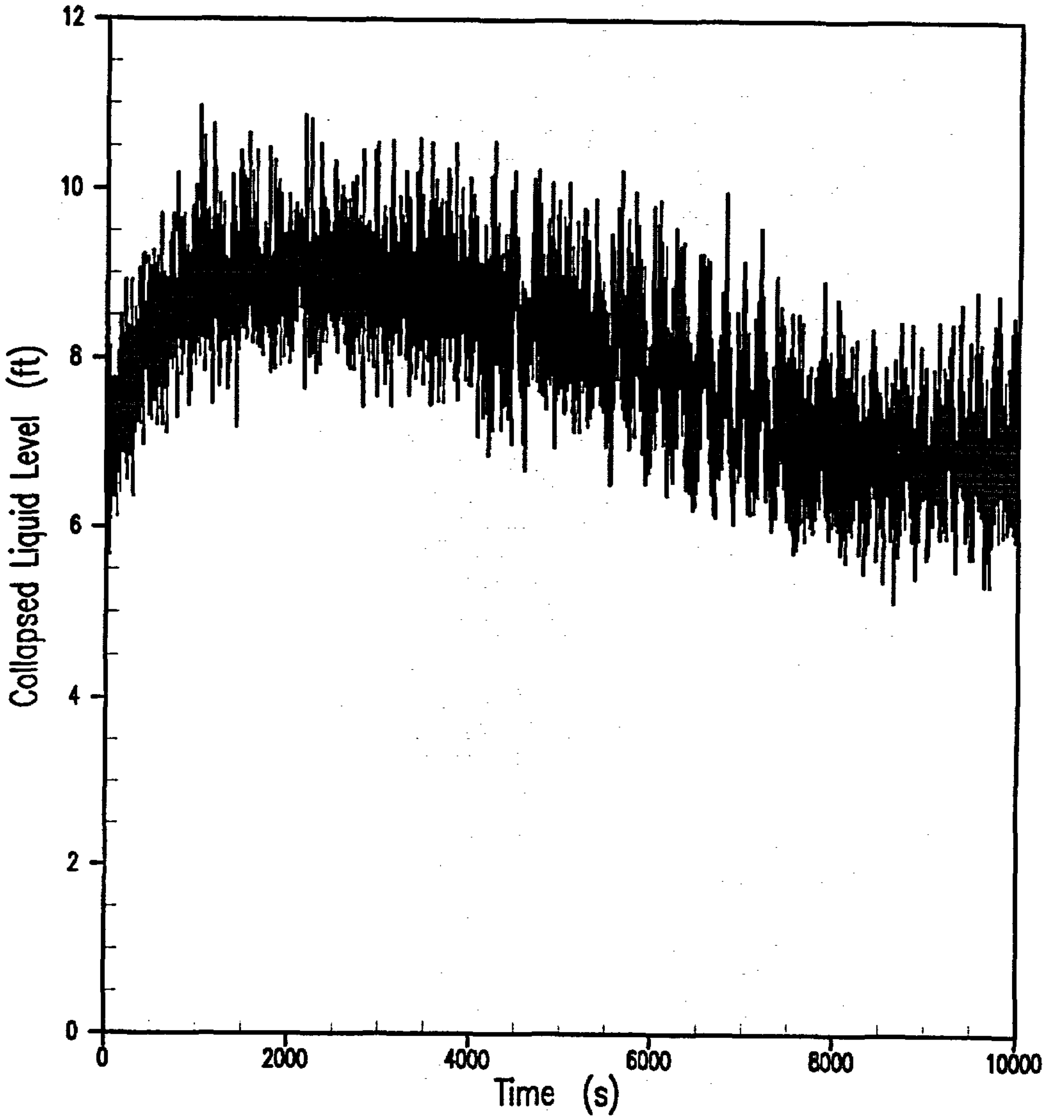

Figure 15.6.5.4C-2

Collapsed Level of Liquid over the Heated Length of the Fuel (DEDVI Case)

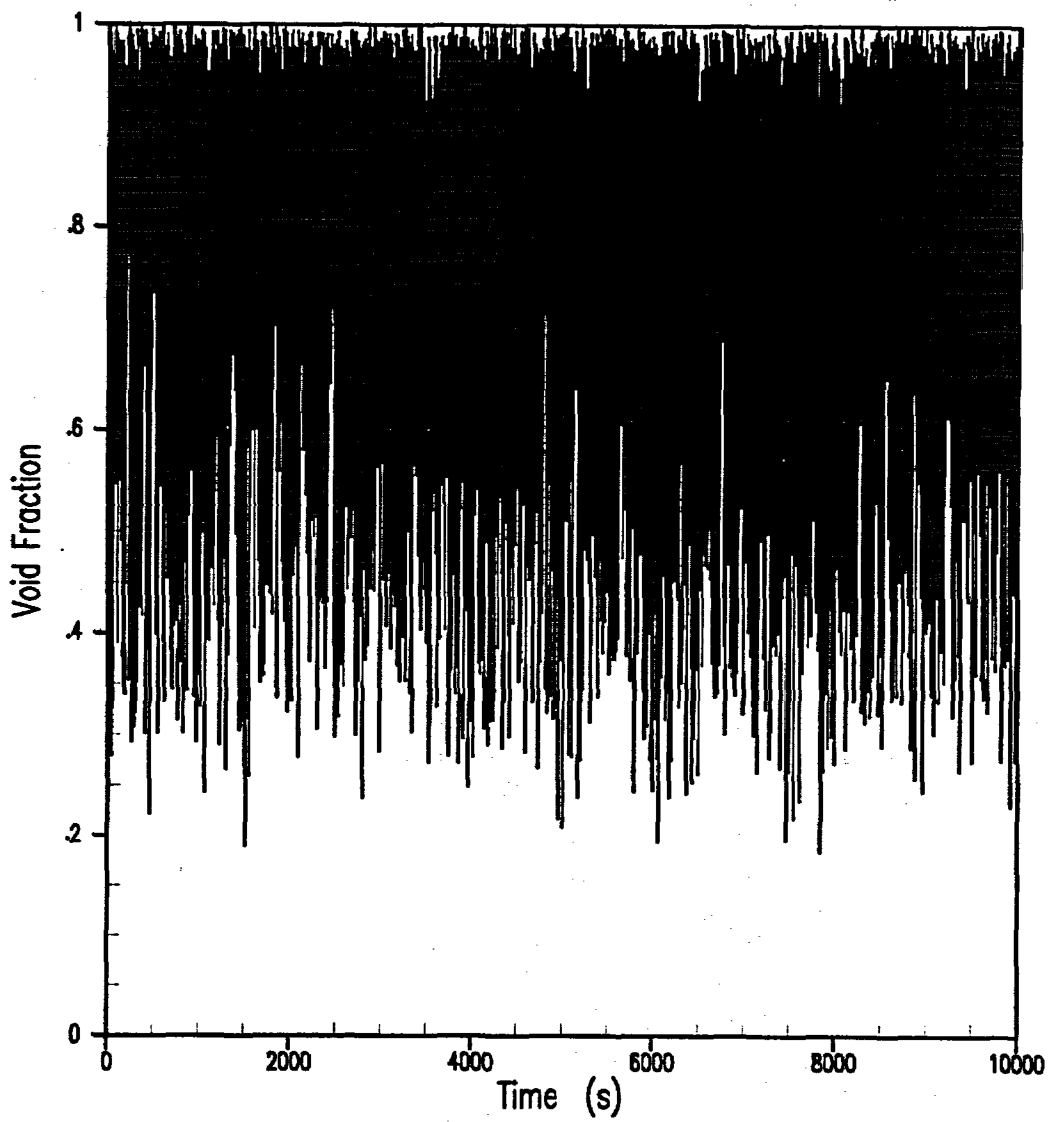

Figure 15.6.5.4C-3

Void Fraction in Core ~~Cell Level 1 of 2~~Hot Assembly Top Cell (DEDVI Case)

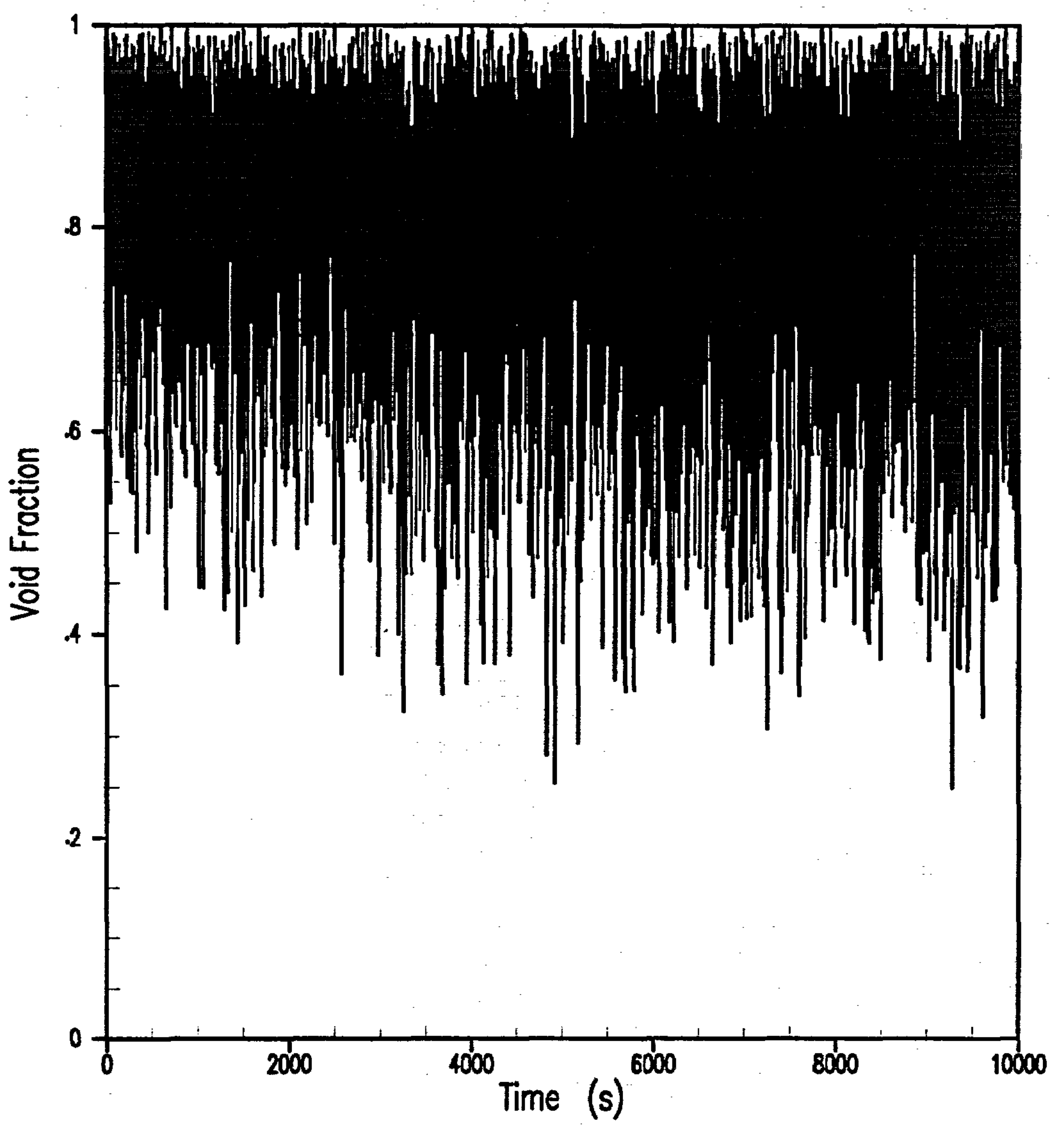

Figure 15.6.5.4C-4

Void Fraction in Core ~~Cell Level 2 of 2~~Hot Assembly Second from Top Cell (DEDVI Case)

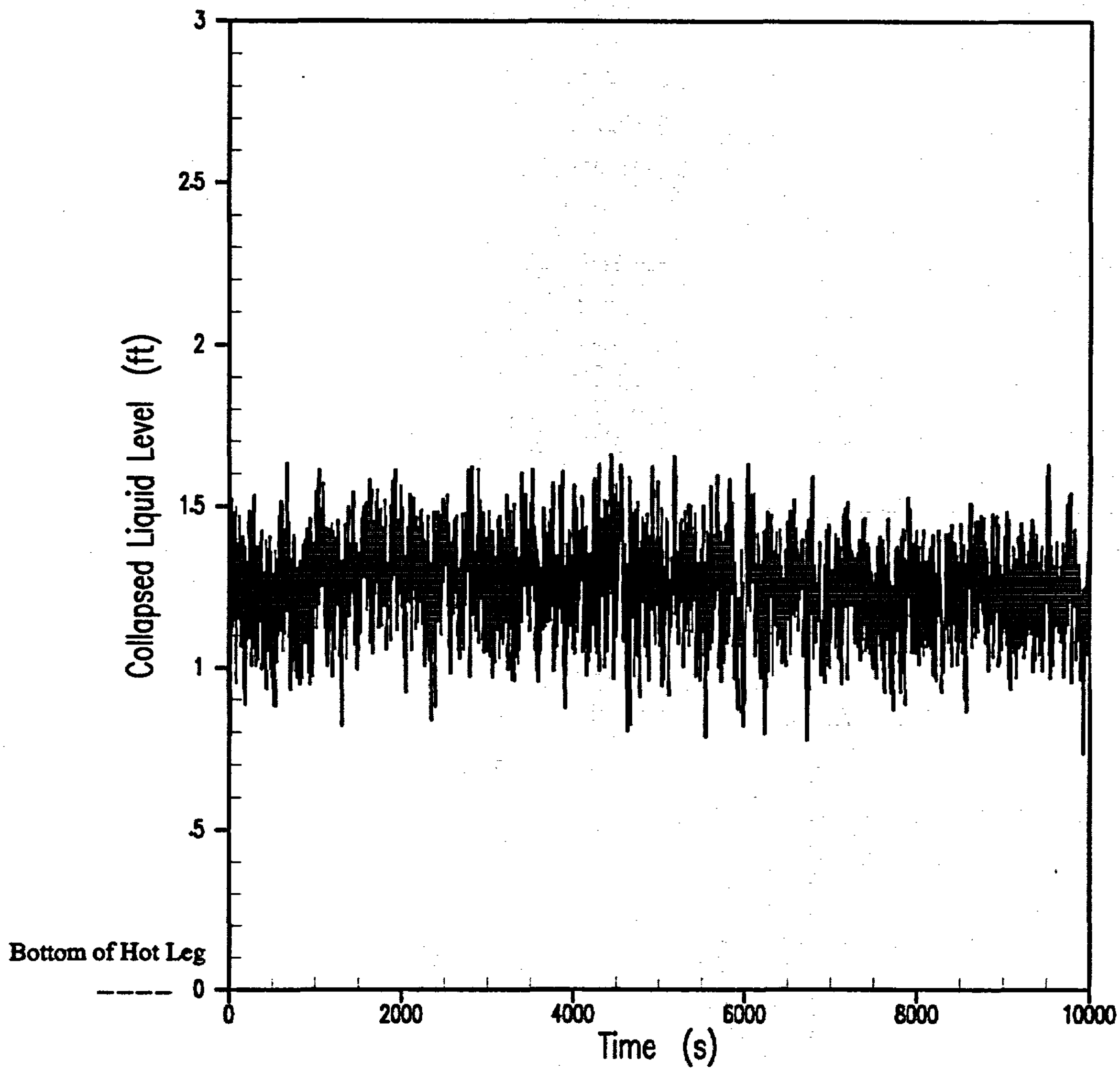

Figure 15.6.5.4C-5

Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (DEDVI Case)

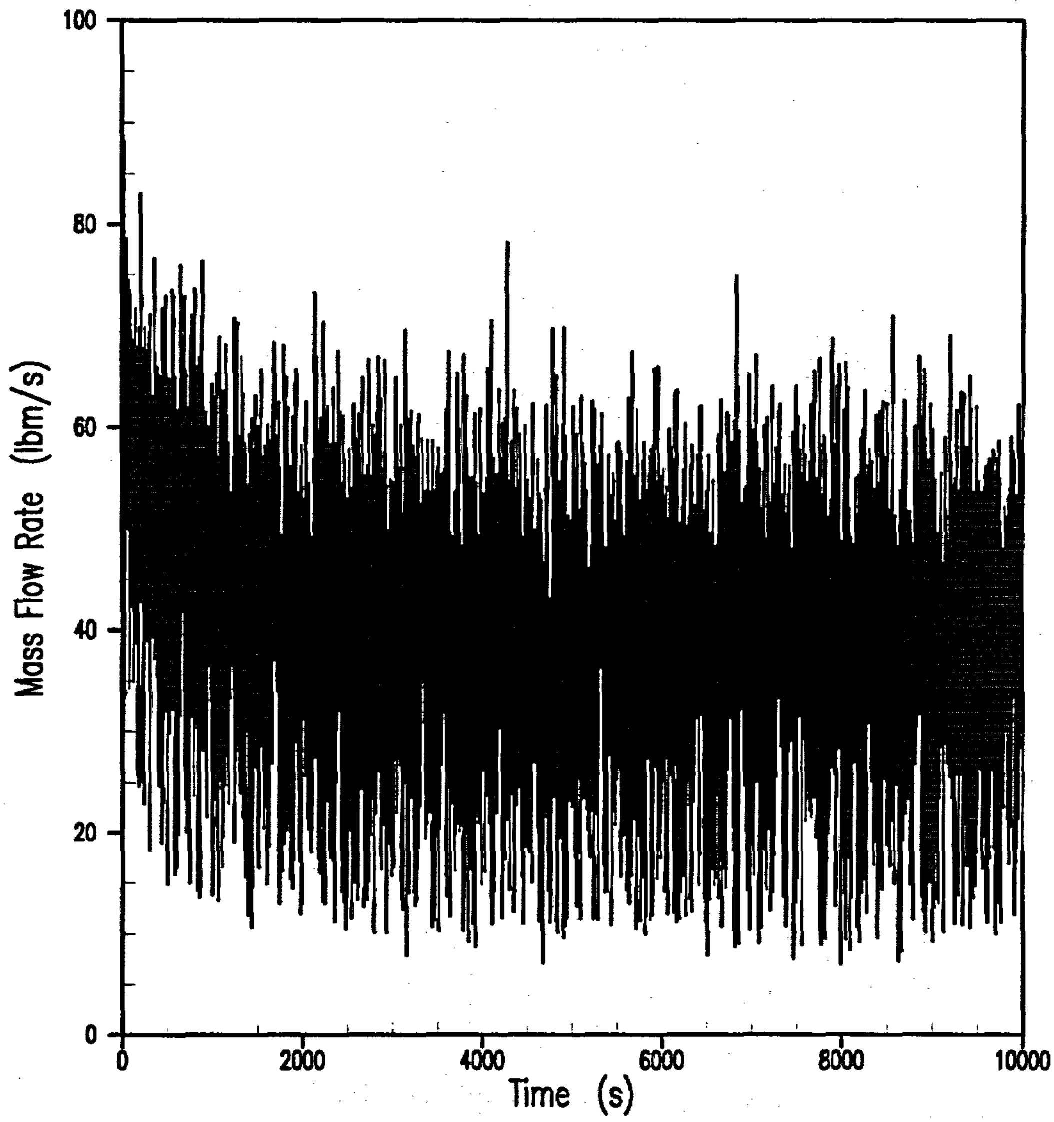

Figure 15.6.5.4C-6

Vapor Rate out of the Core (DEDVI Case)

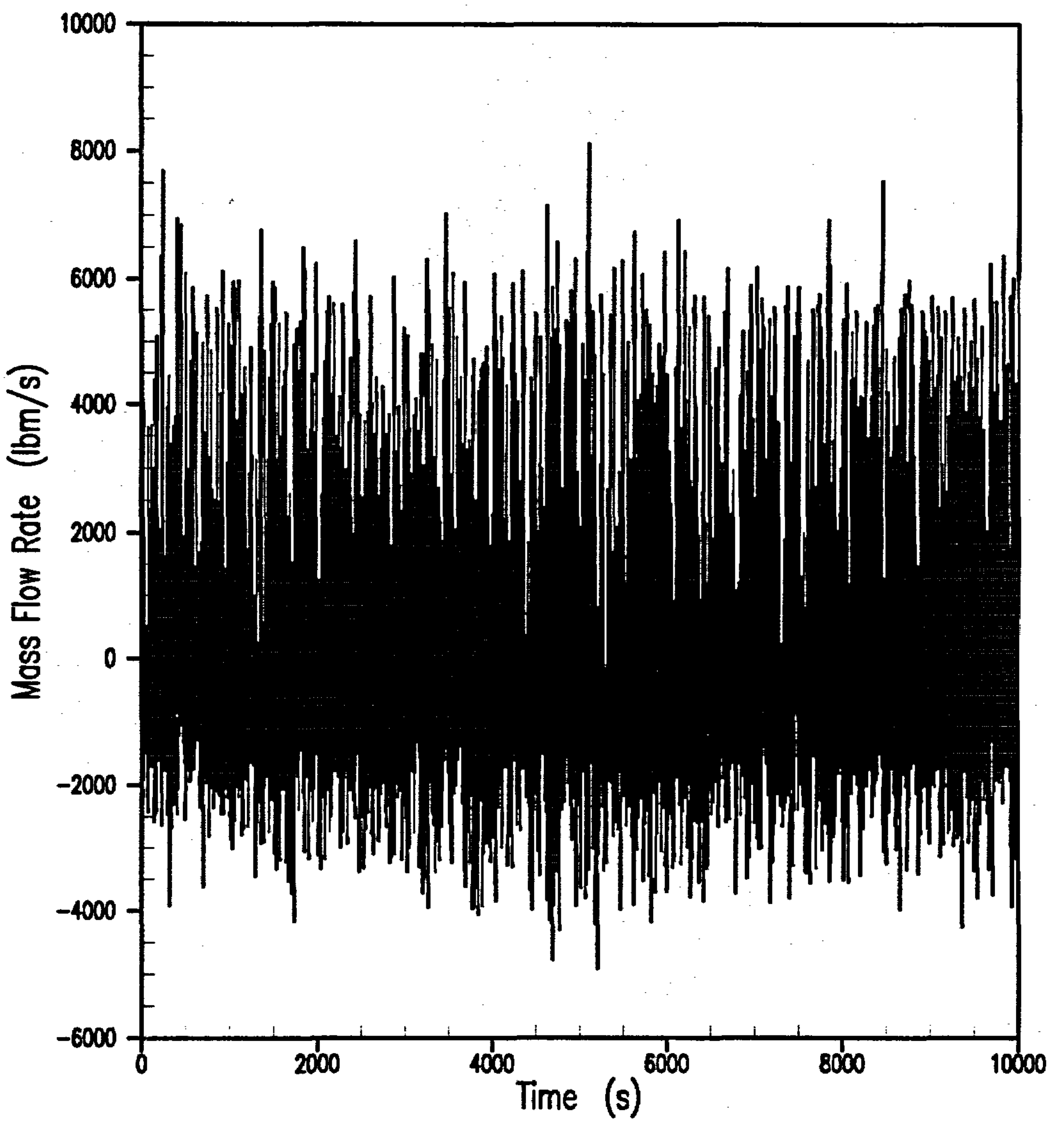

Figure 15.6.5.4C-7

Liquid Flow Rate out of the Core (DEDVI Case)

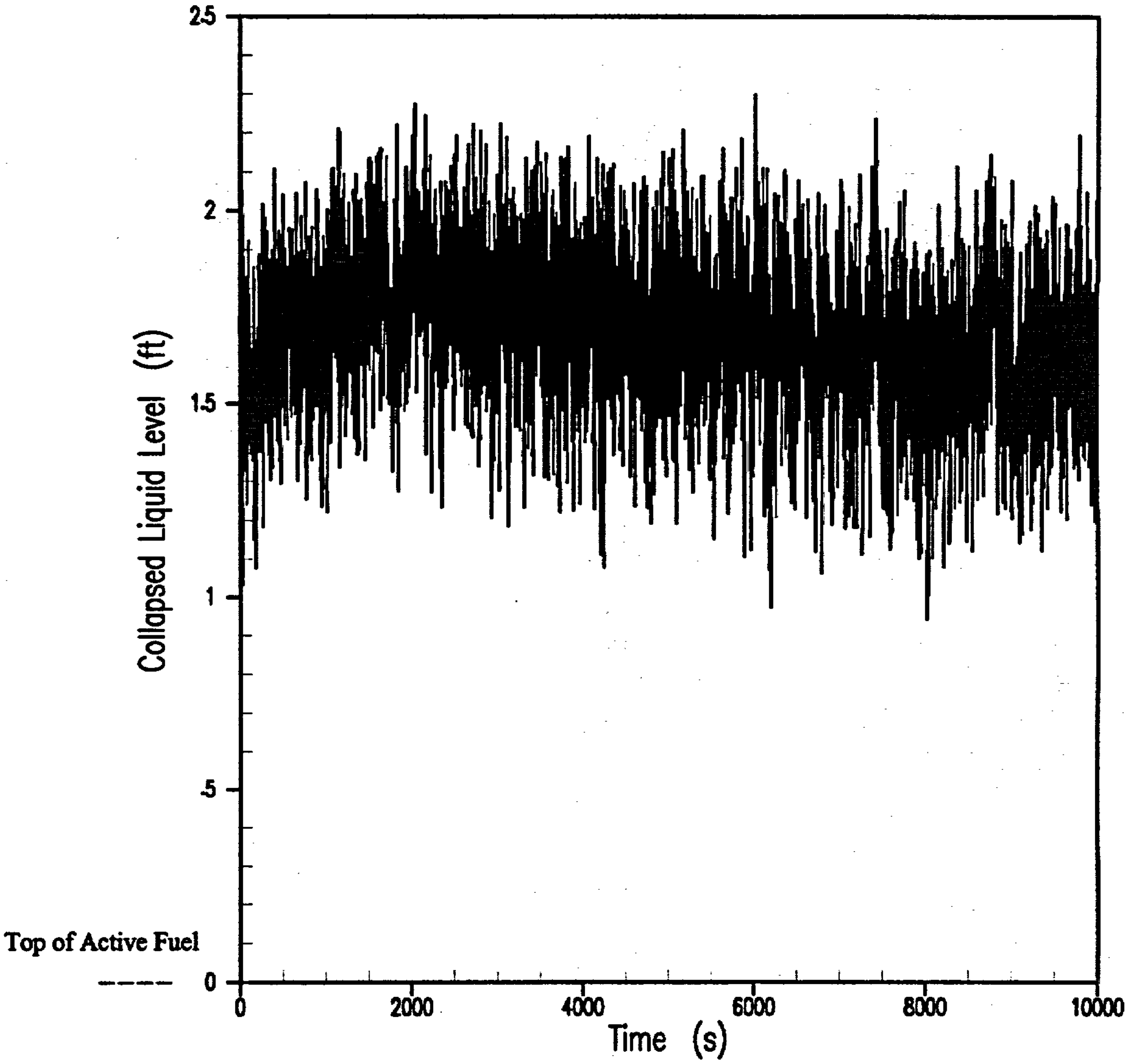

Figure 15.6.5.4C-8

Collapsed Liquid Level in the Upper Plenum (DEDVI Case)

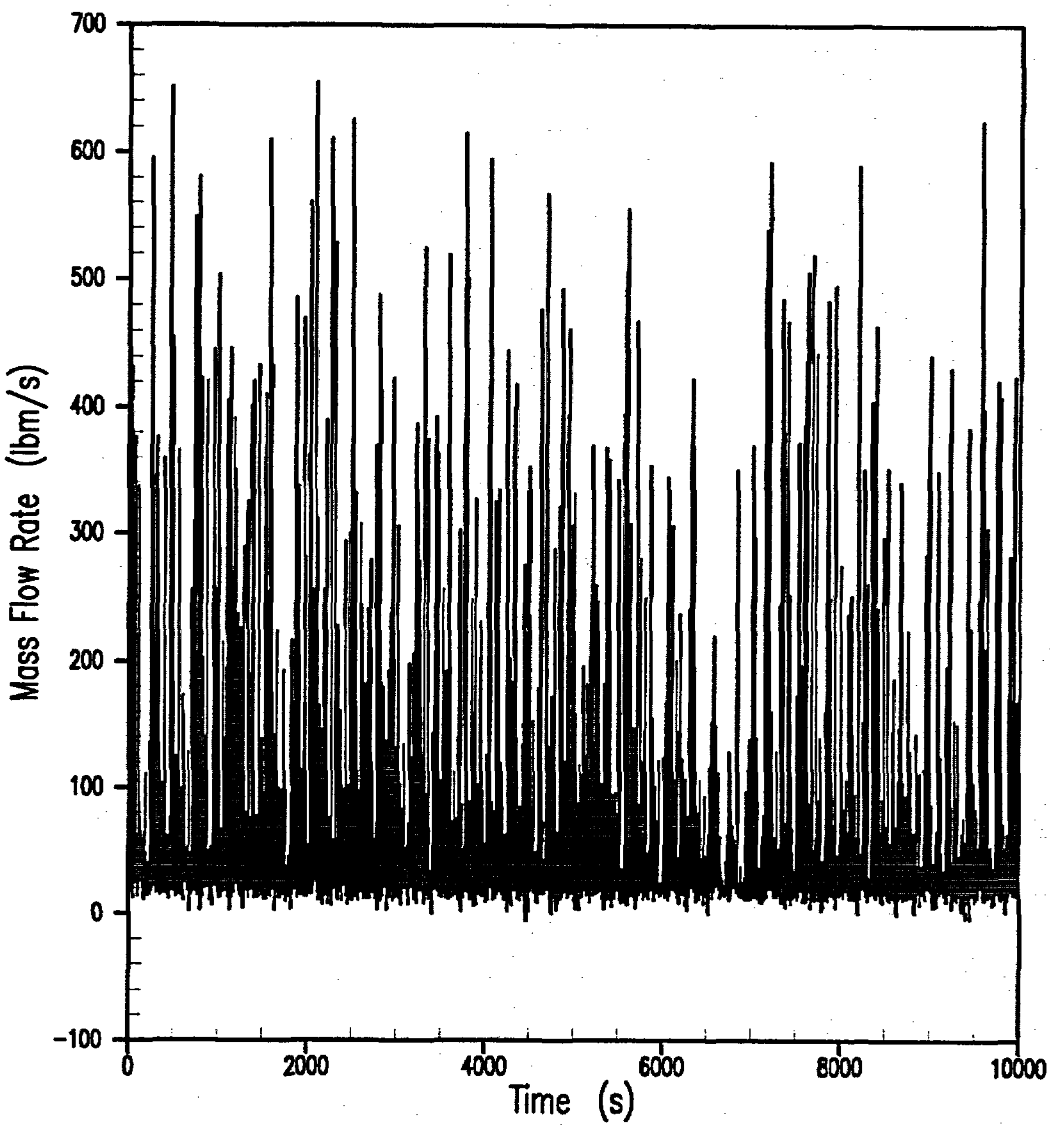

Figure 15.6.5.4C-9

Mixture Flow Rate Through ADS Stage 4A Valves (DEDVI Case)

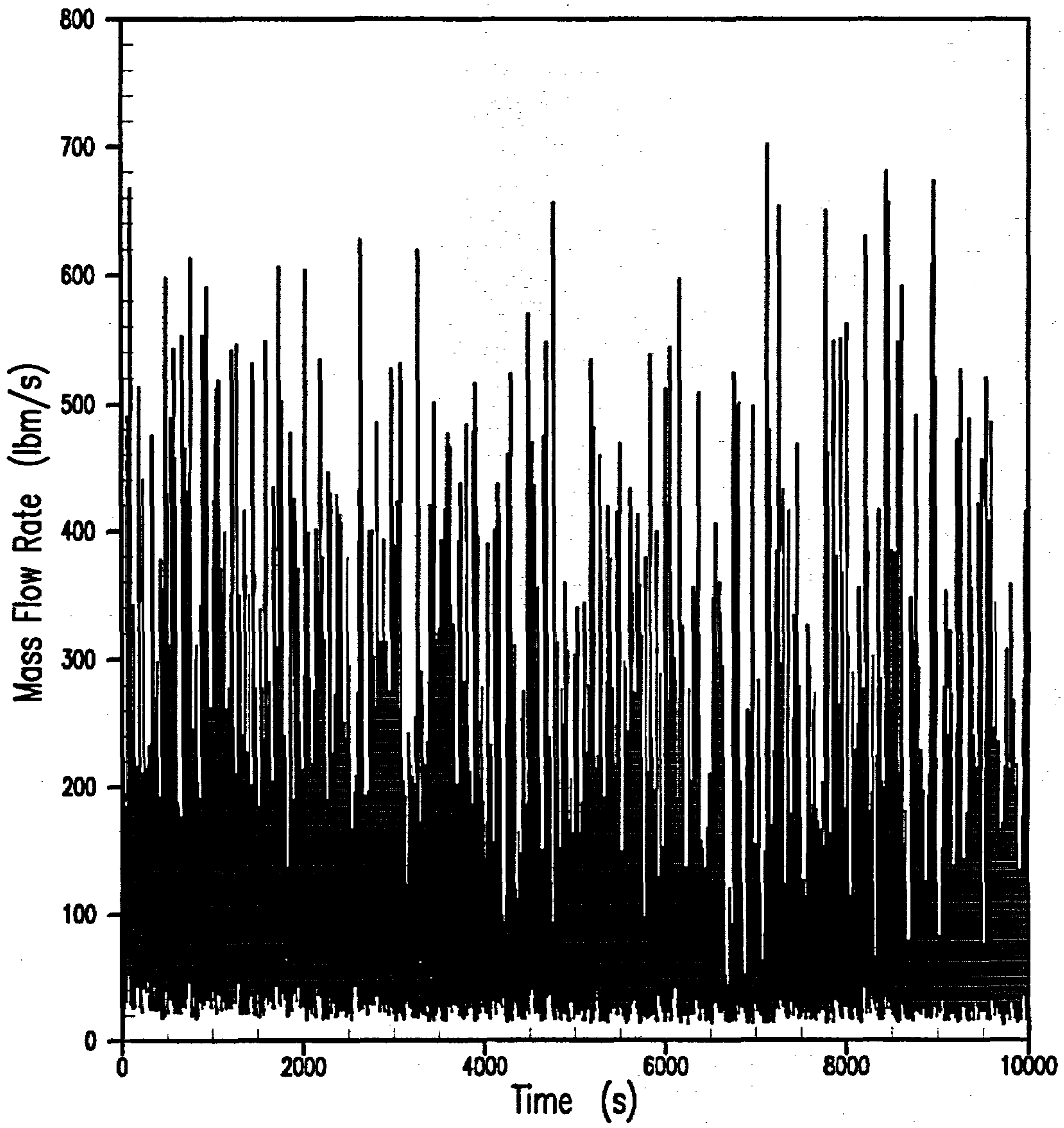

Figure 15.6.5.4C-10

Mixture Flow Rate Through ADS Stage 4B Valves (DEDVI Case)

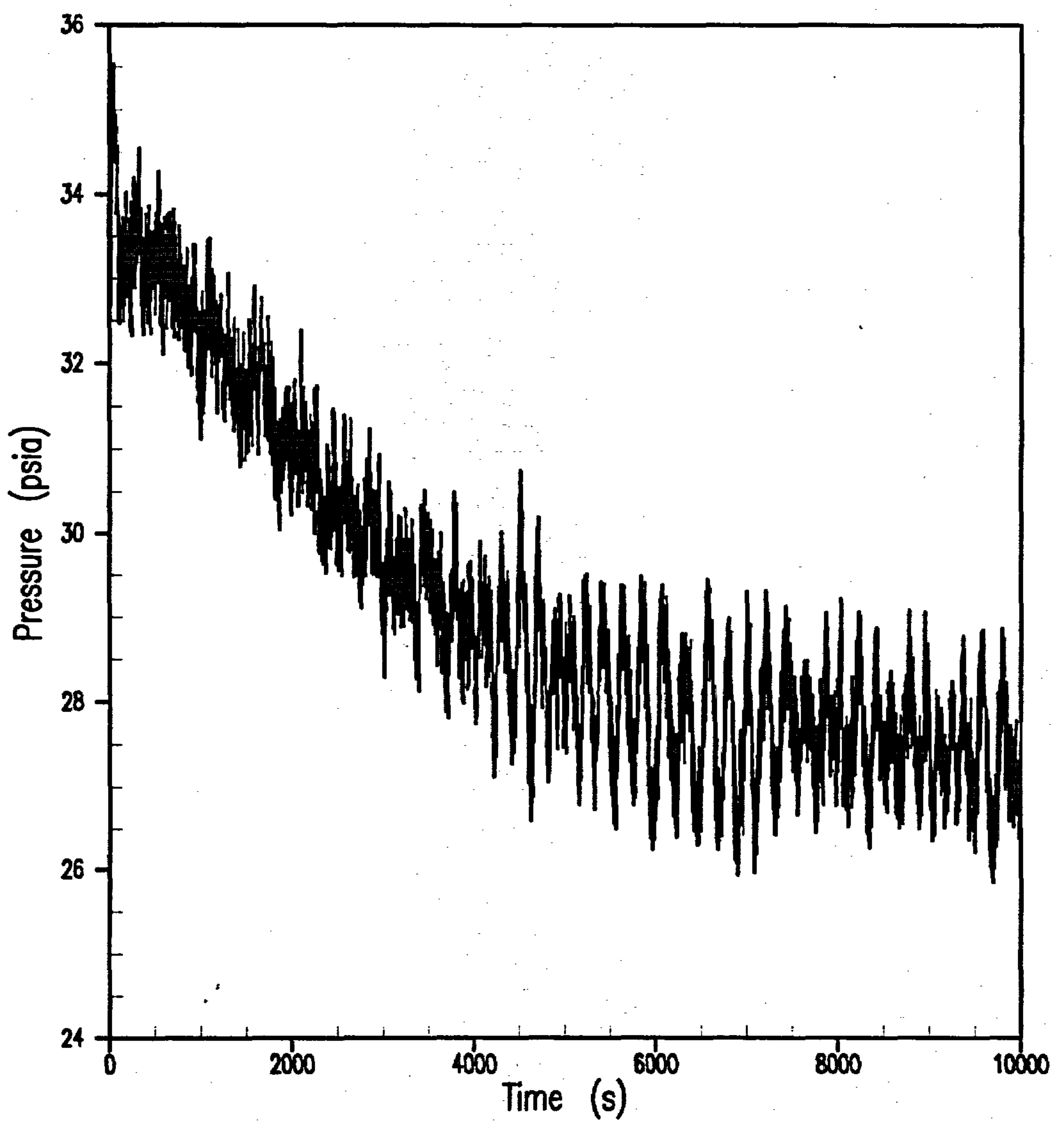

Figure 15.6.5.4C-11

Upper Plenum Pressure (DEDVI Case)

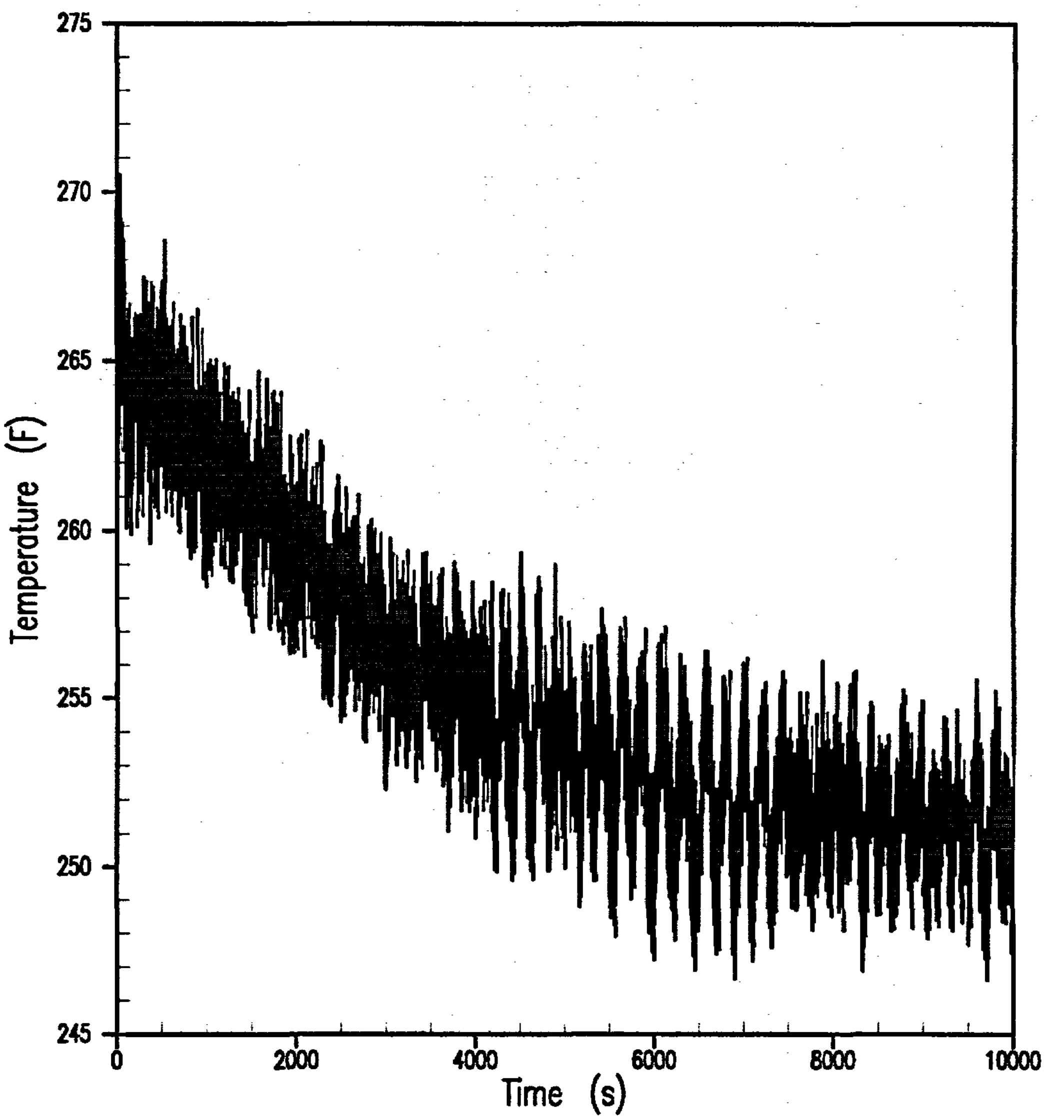

Figure 15.6.5.4C-12

~~PCT of the Hot Rod~~Hot Rod Cladding Temperature Near Top of Core (DEDVI Case)

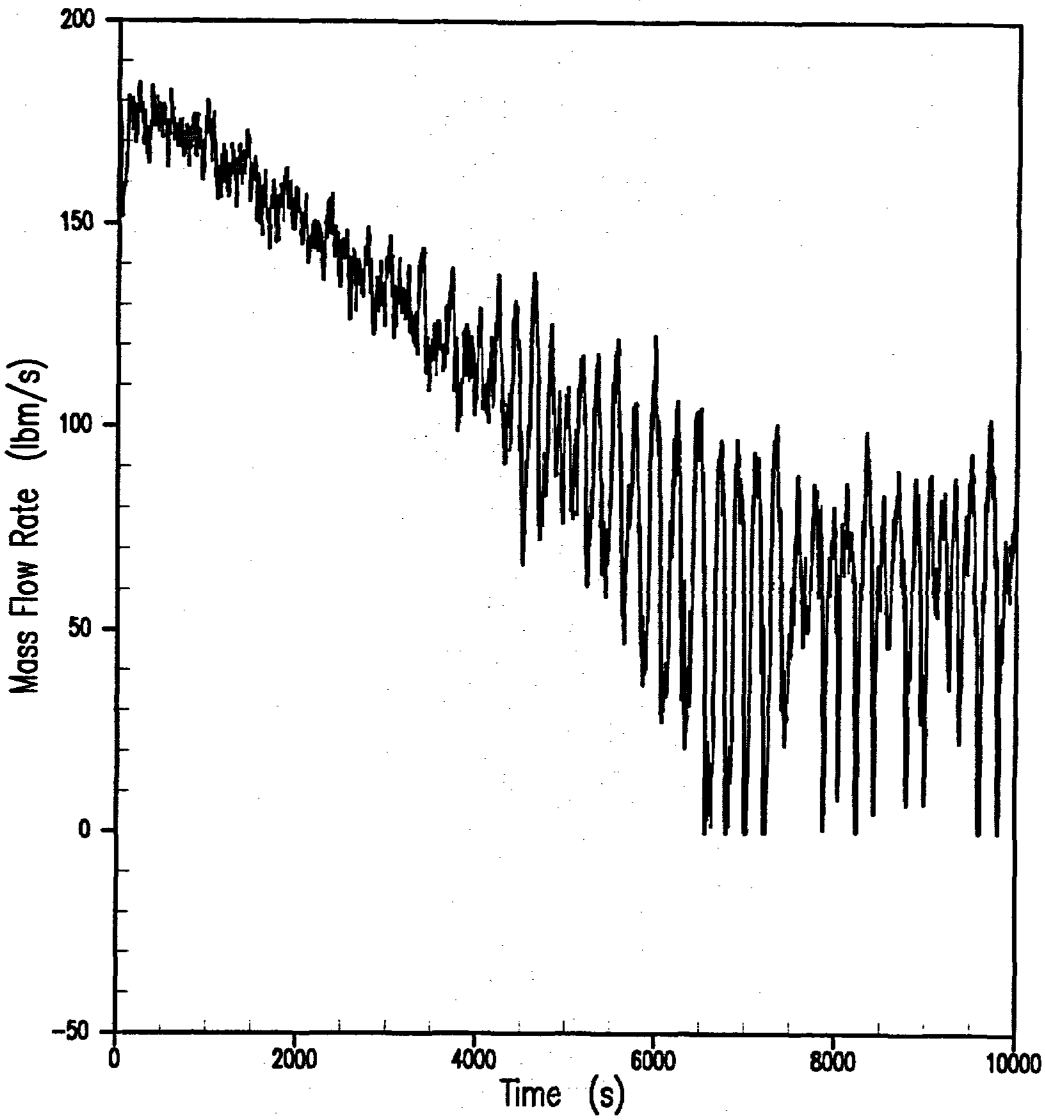

Figure 15.6.5.4C-13

DVI–A Mixture Flow Rate (DEDVI Case)

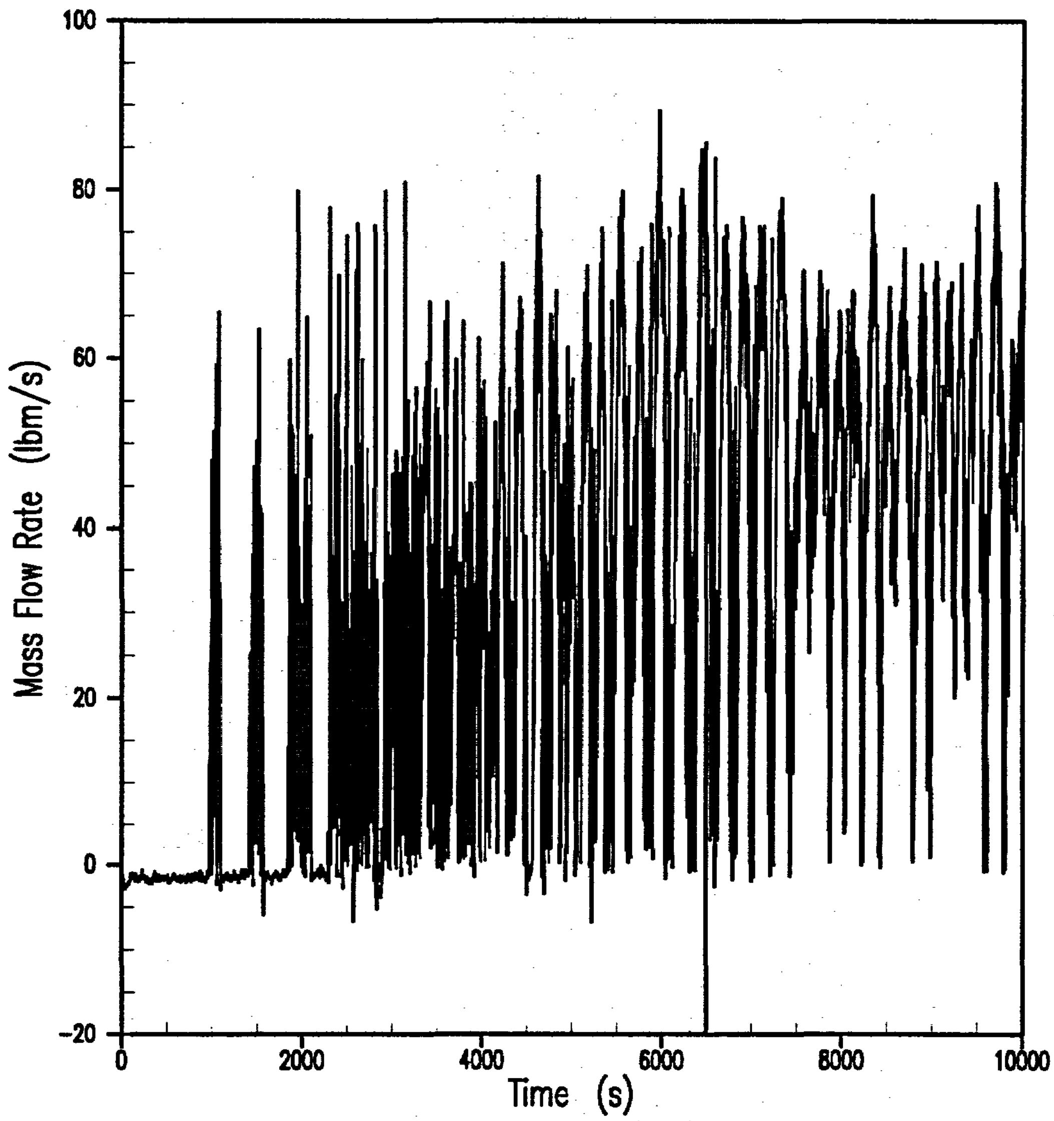

Figure 15.6.5.4C-14

DVI–B Mixture Flow Rate (DEDVI Case)

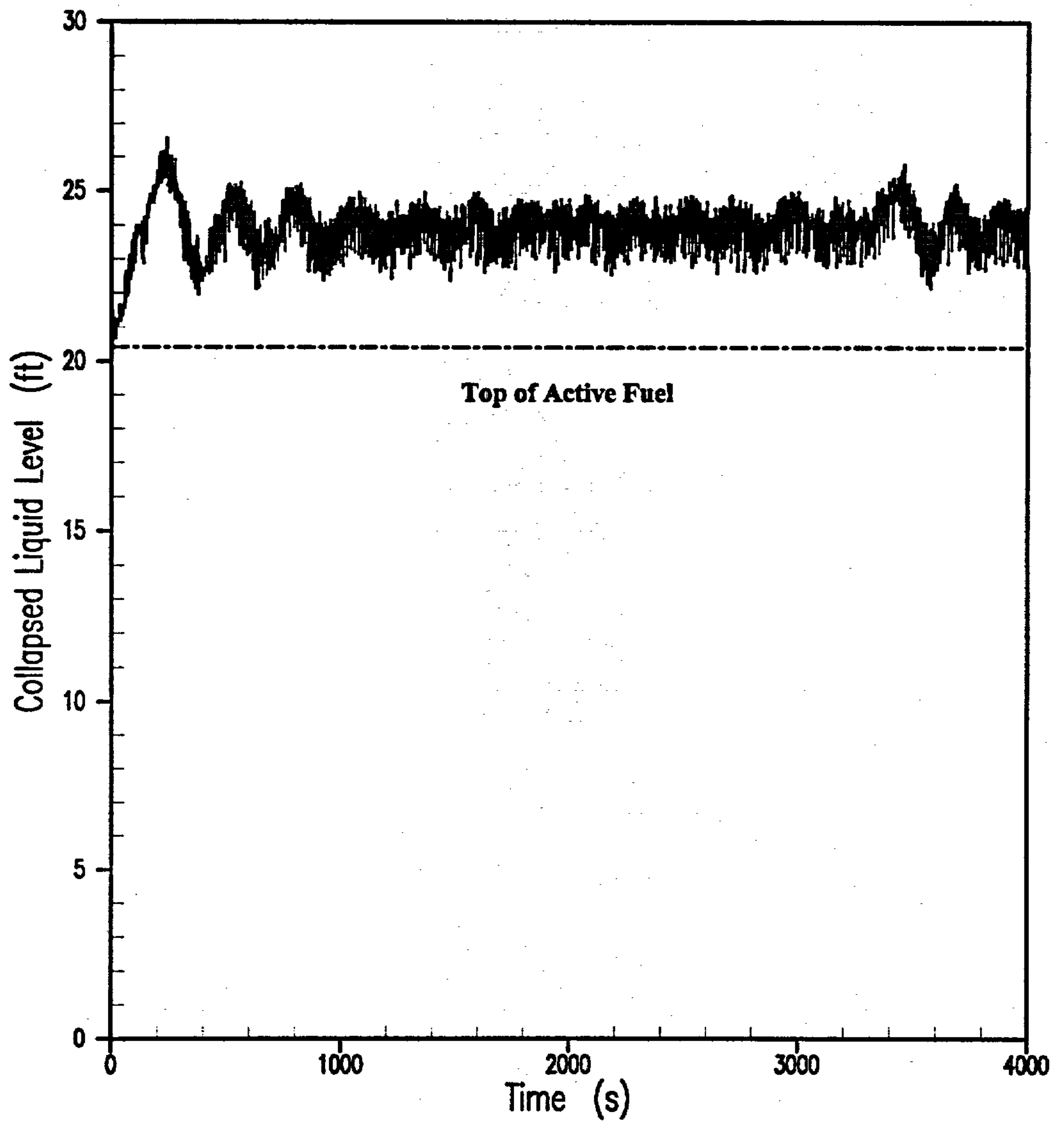

Figure 15.6.5.4C-15

Collapsed Level of Liquid in the Downcomer
(Wall-to-Wall Floodup Case)

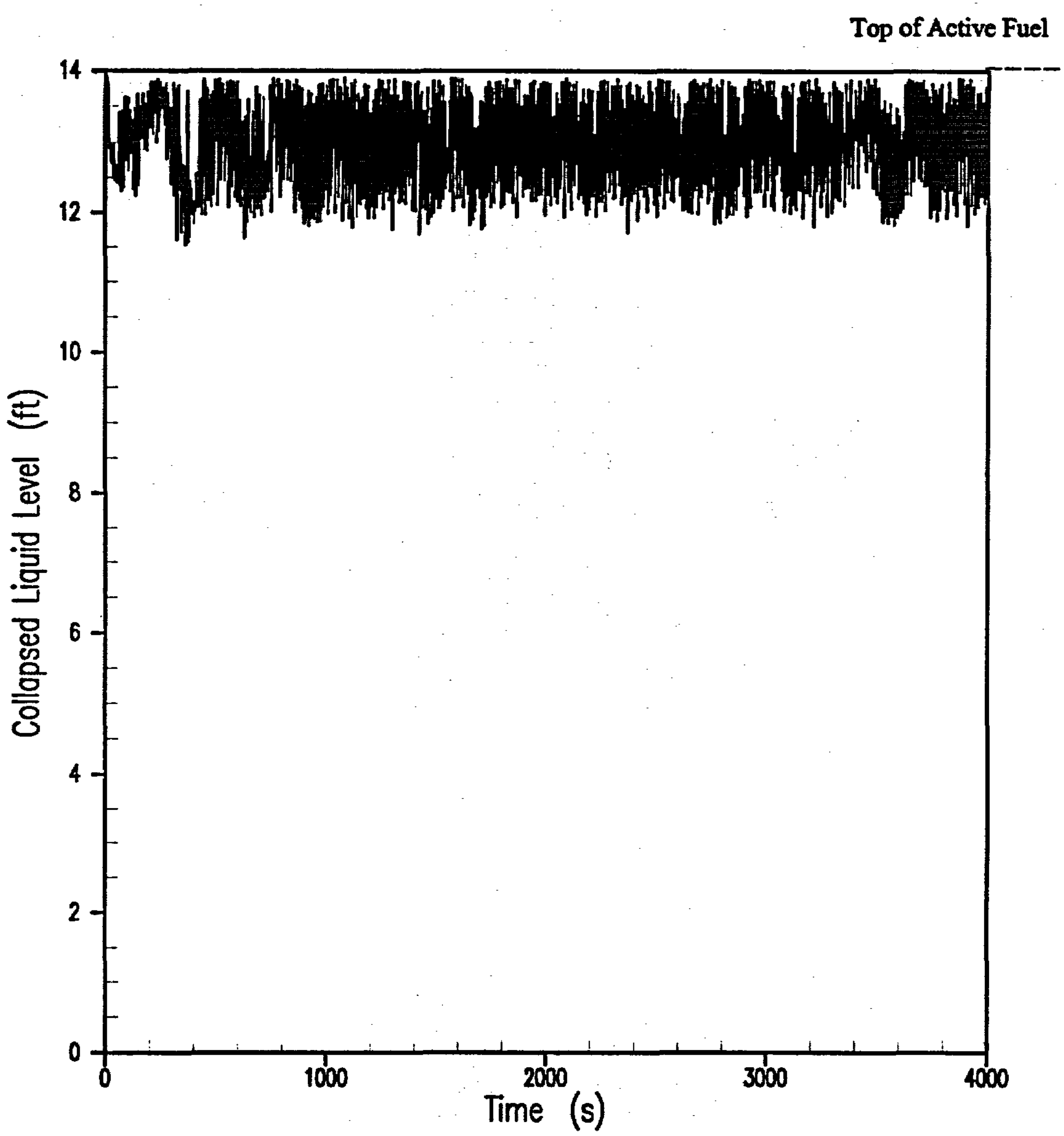

Figure 15.6.5.4C-16

Collapsed Level of Liquid Over the Heated Length of the Fuel (Wall-to-Wall Floodup Case)

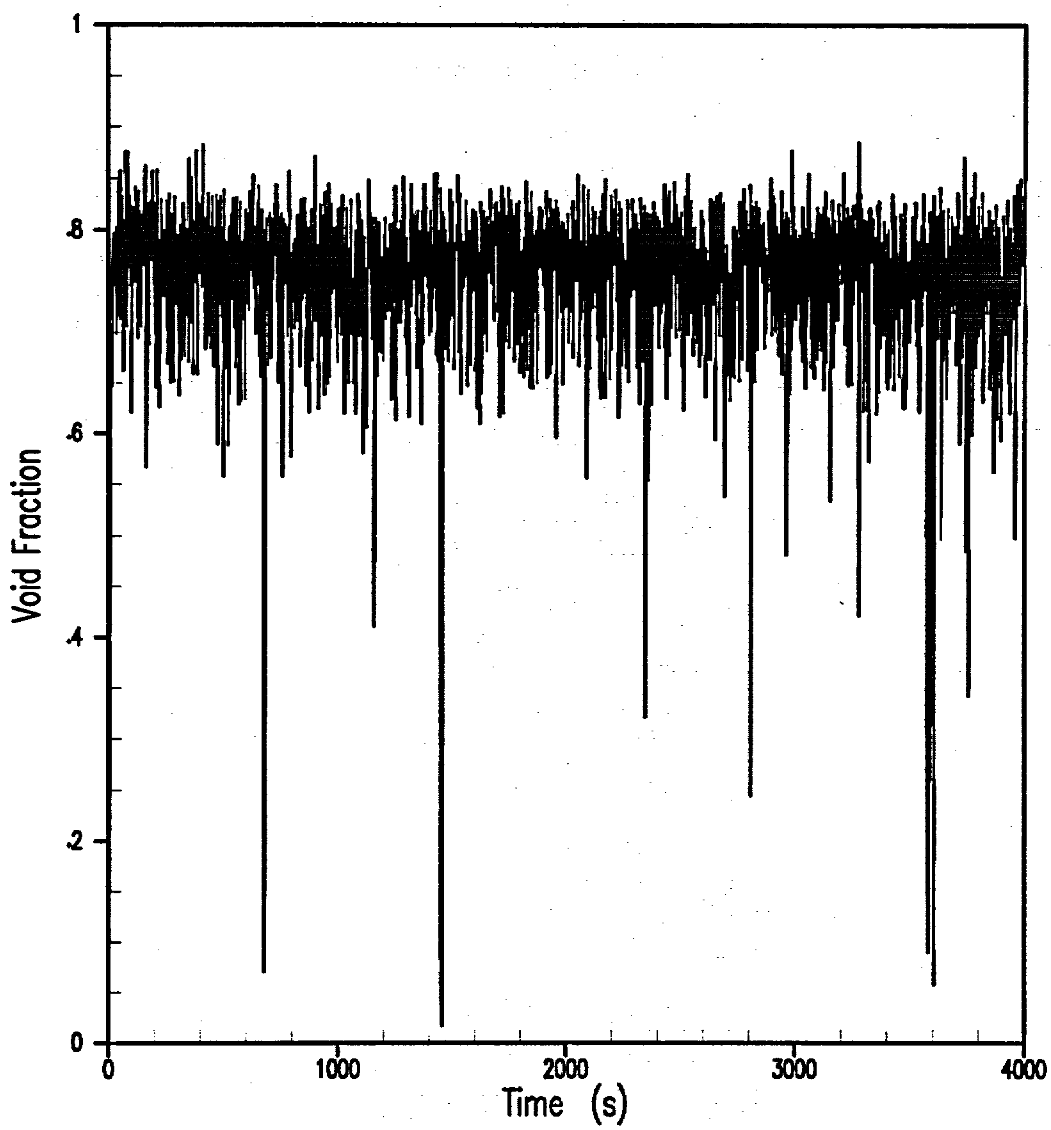

Figure 15.6.5.4C-17

Void Fraction in Core ~~Cell Level 1 of 2~~Hot Assembly Top Cell (Wall-to-Wall Floodup Case)

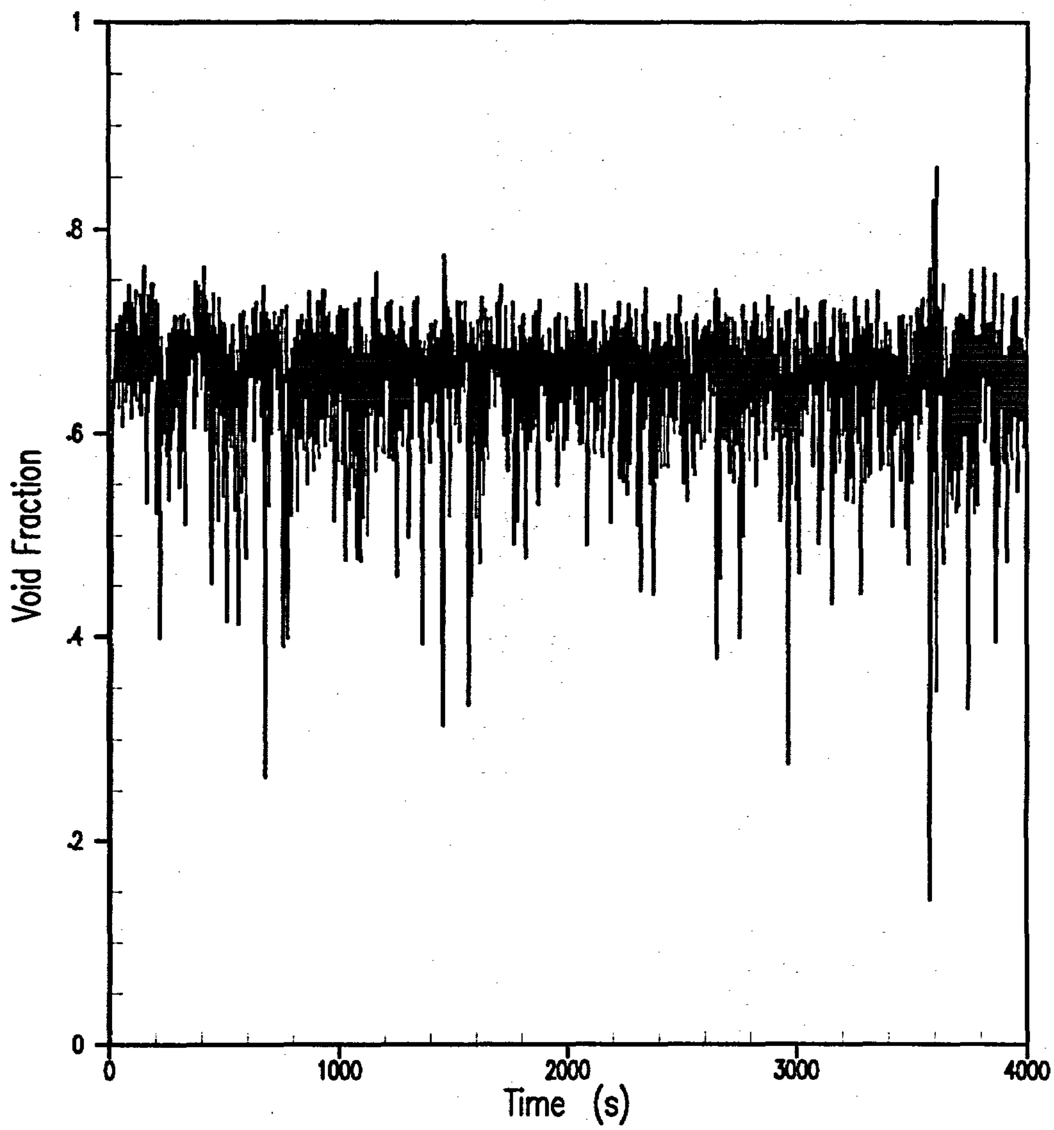

Figure 15.6.5.4C-18

Void Fraction in Core ~~Cell Level 2 of 2~~Hot Assembly Second from Top Cell (Wall-to-Wall Floodup Case)

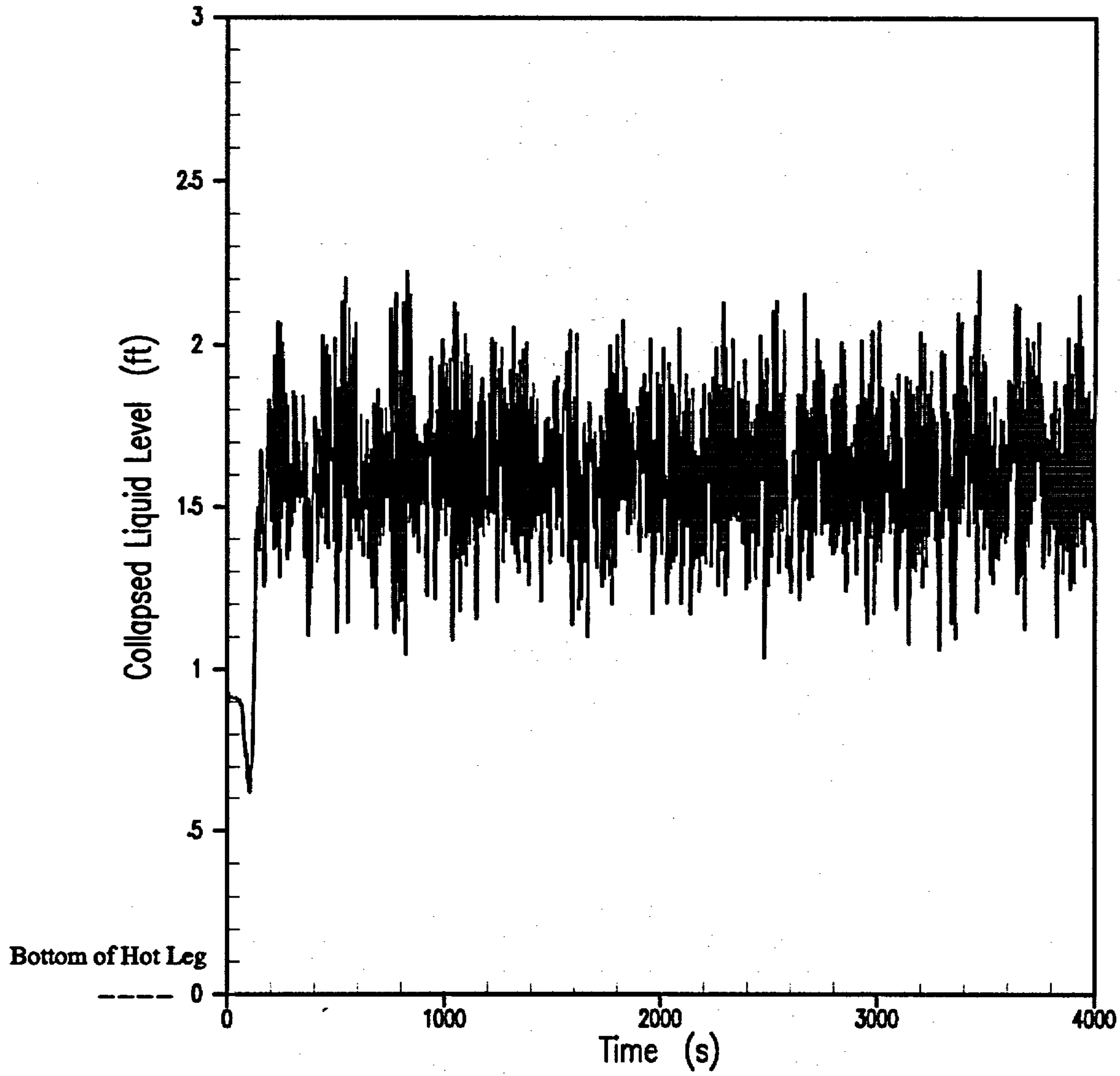

Figure 15.6.5.4C-19

Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (Wall-to-Wall Floodup Case)

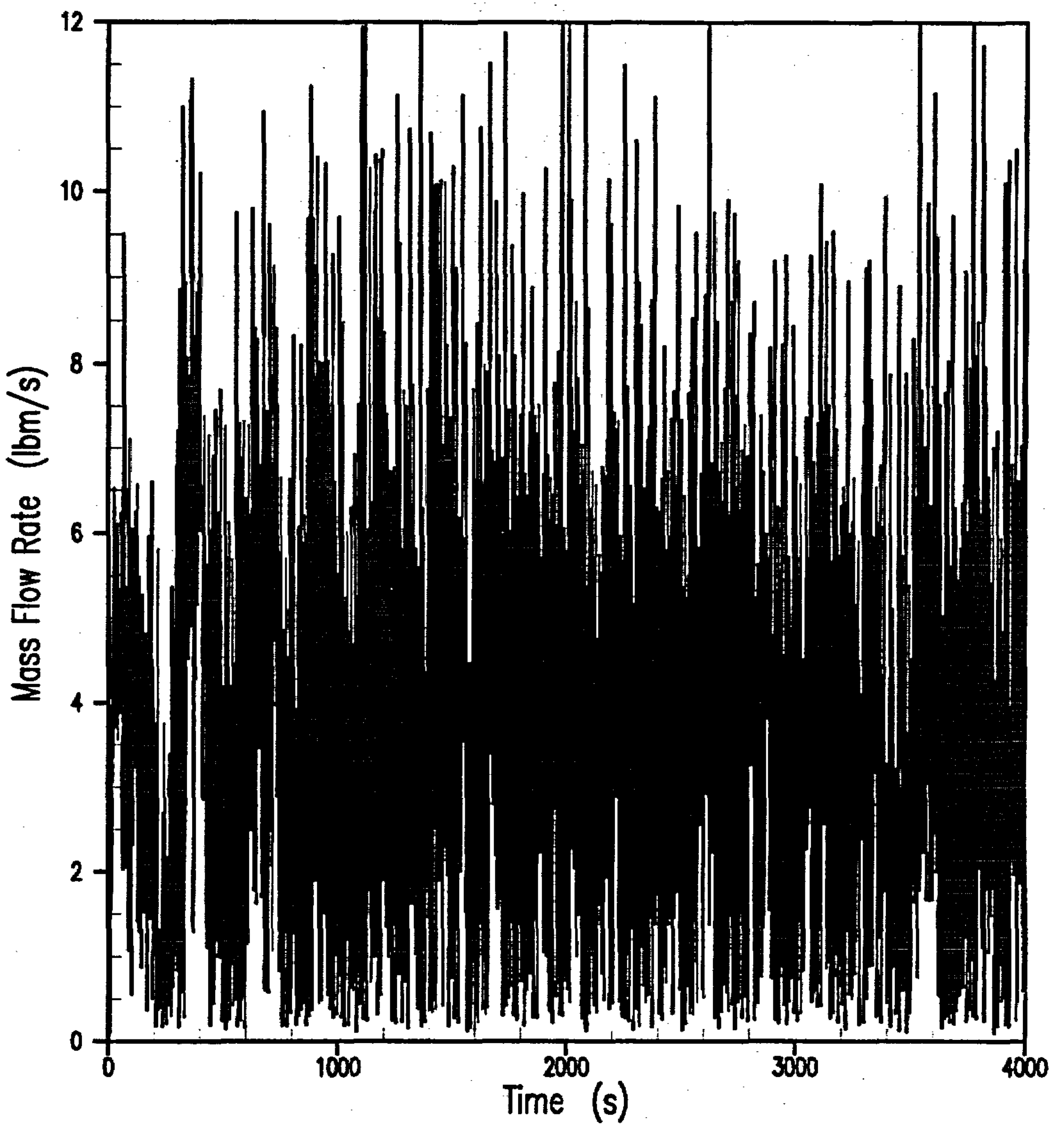

Figure 15.6.5.4C-20

Vapor Rate out of the Core
(Wall-to-Wall Floodup Case)

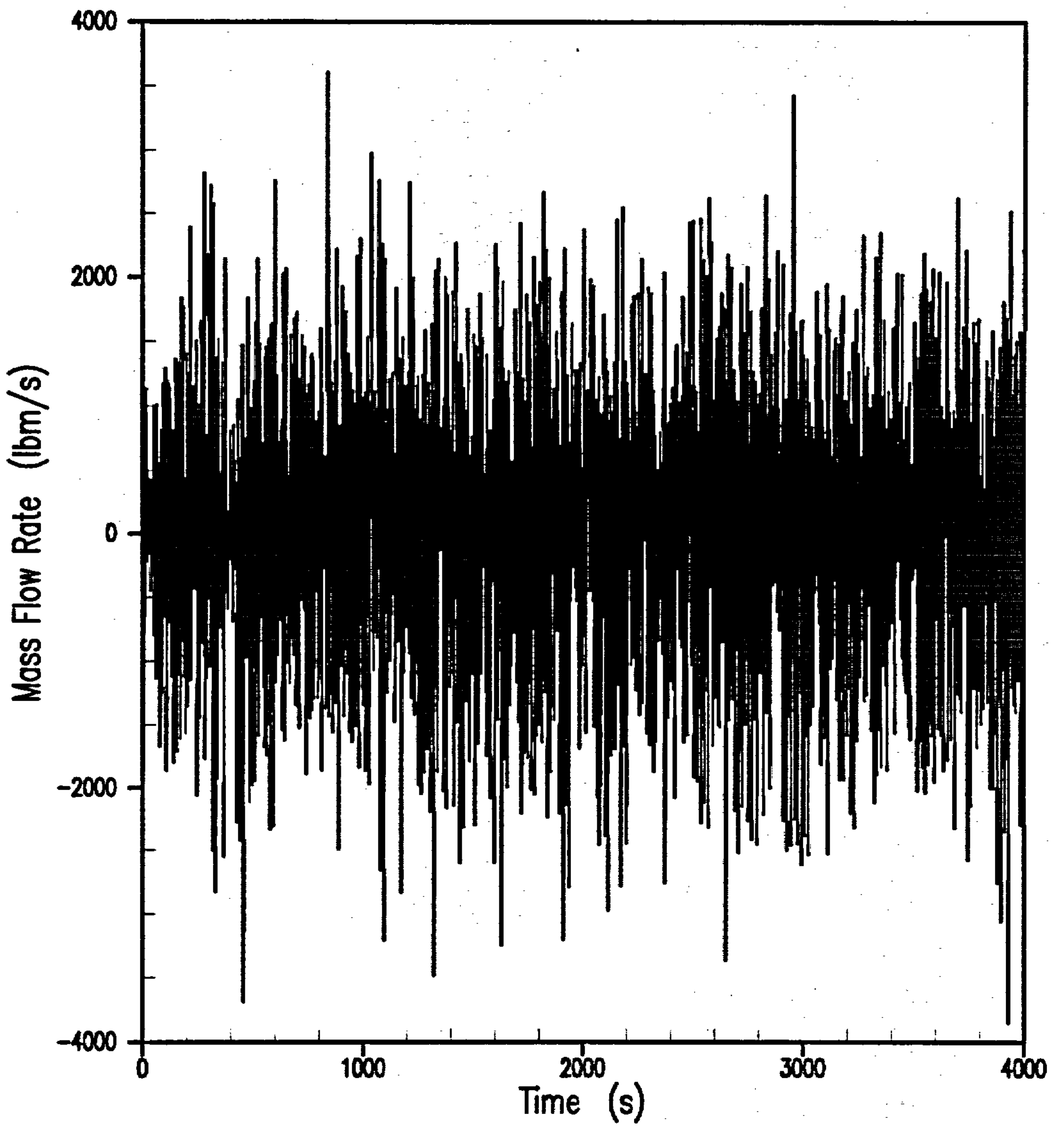

Figure 15.6.5.4C-21

Liquid Flow Rate out of the Core (Wall-to-Wall Floodup Case)

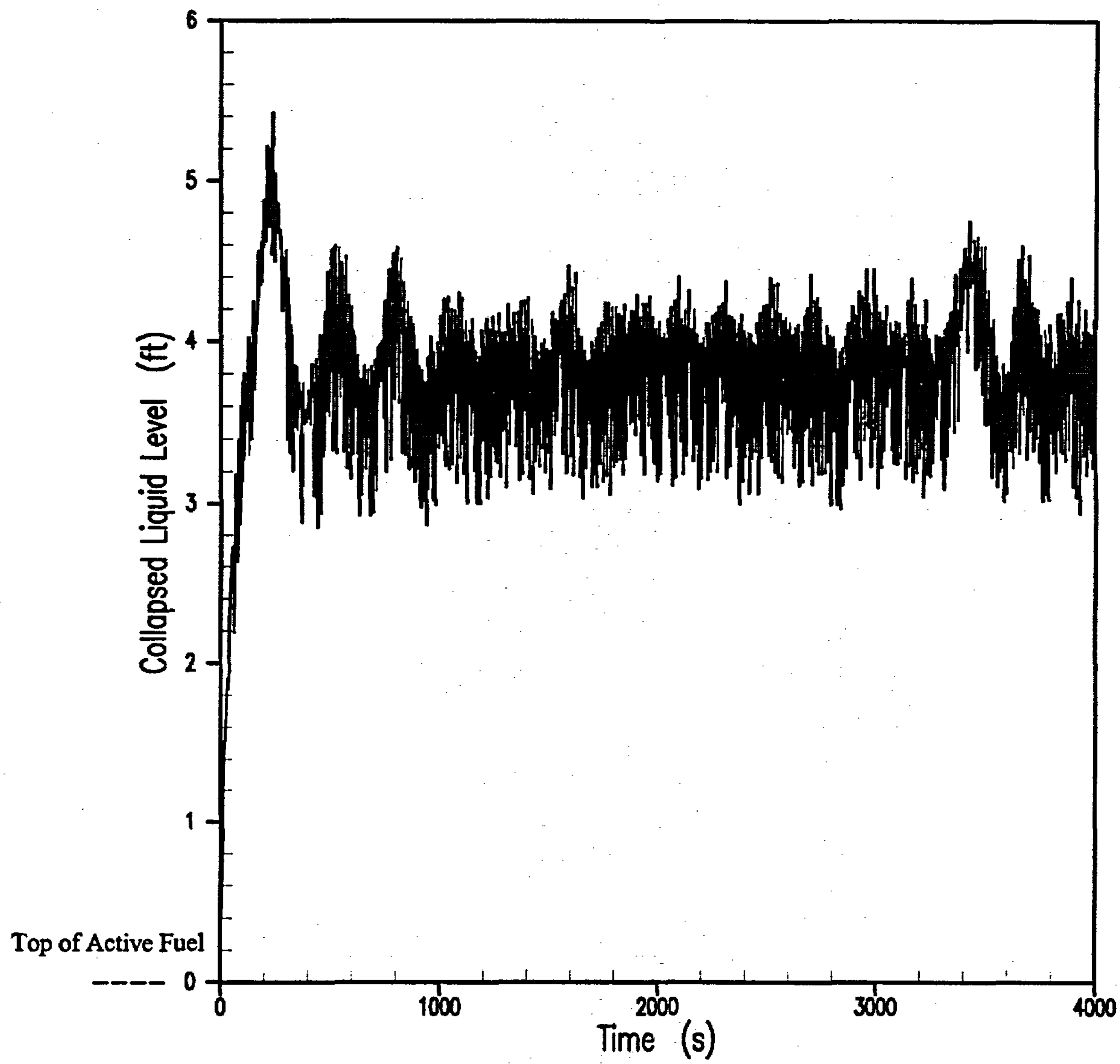

Figure 15.6.5.4C-22

Collapsed Liquid Level in the Upper Plenum (Wall-to-Wall Floodup Case)

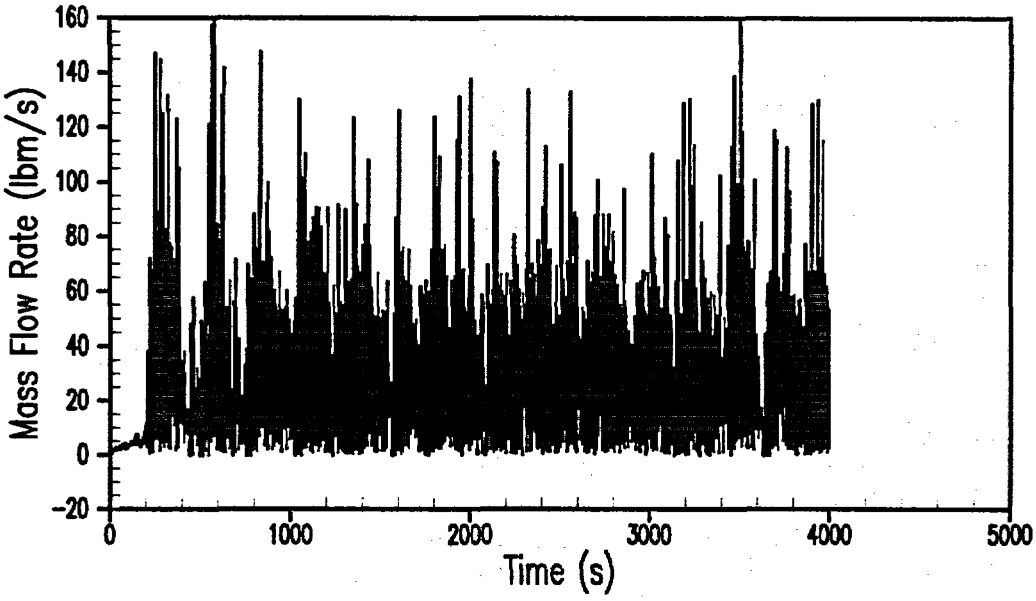

Figure 15.6.5.4C-23

Mixture Flow Rate Through ADS Stage 4A Valves (Wall-to-Wall Floodup Case)

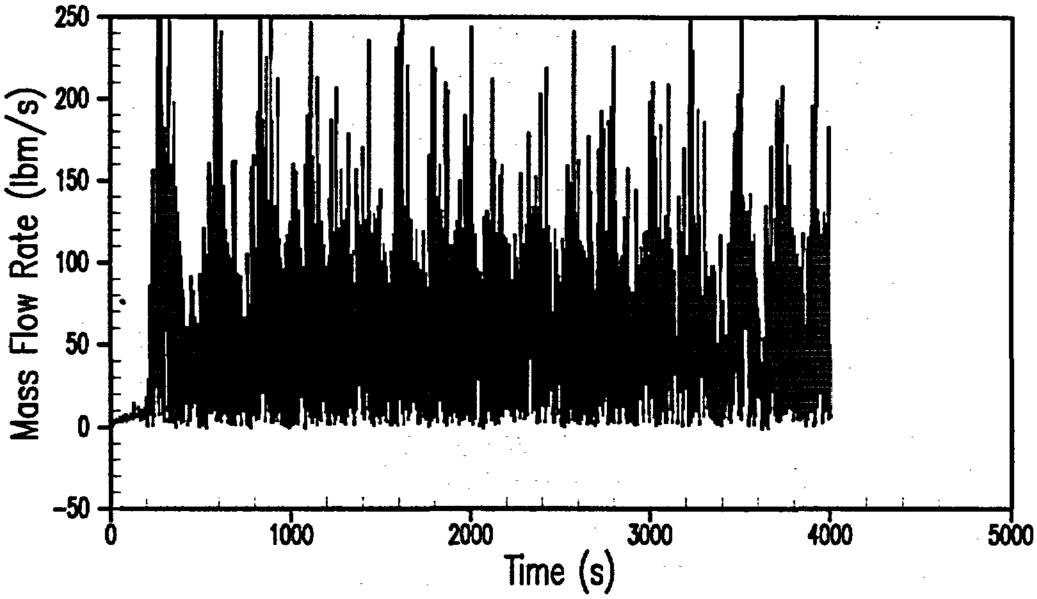

Figure 15.6.5.4C-24

Mixture Flow Rate Through ADS Stage 4B Valves (Wall-to-Wall Floodup Case)

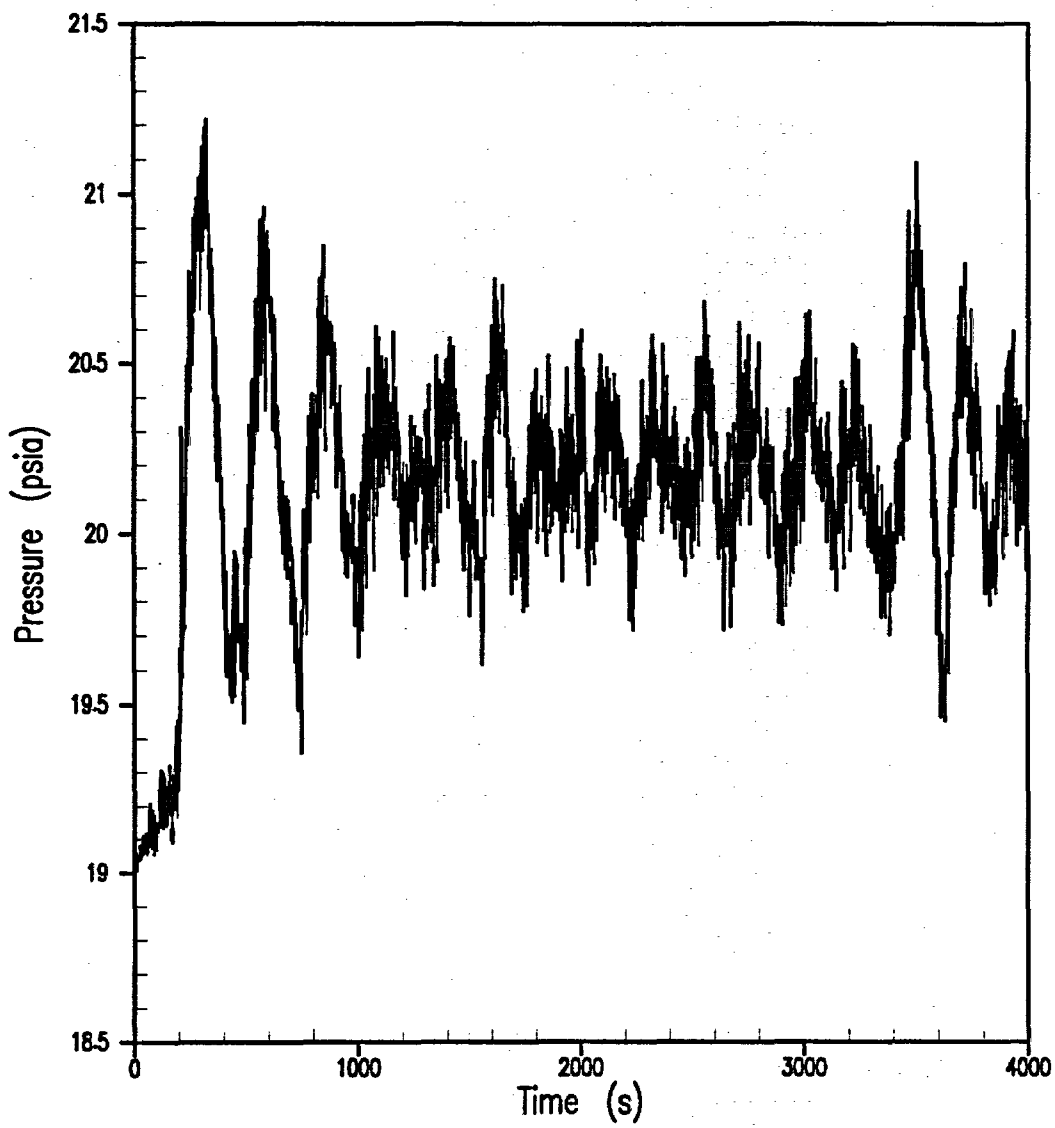

Figure 15.6.5.4C-25

Upper Plenum Pressure
(Wall-to-Wall Floodup Case)

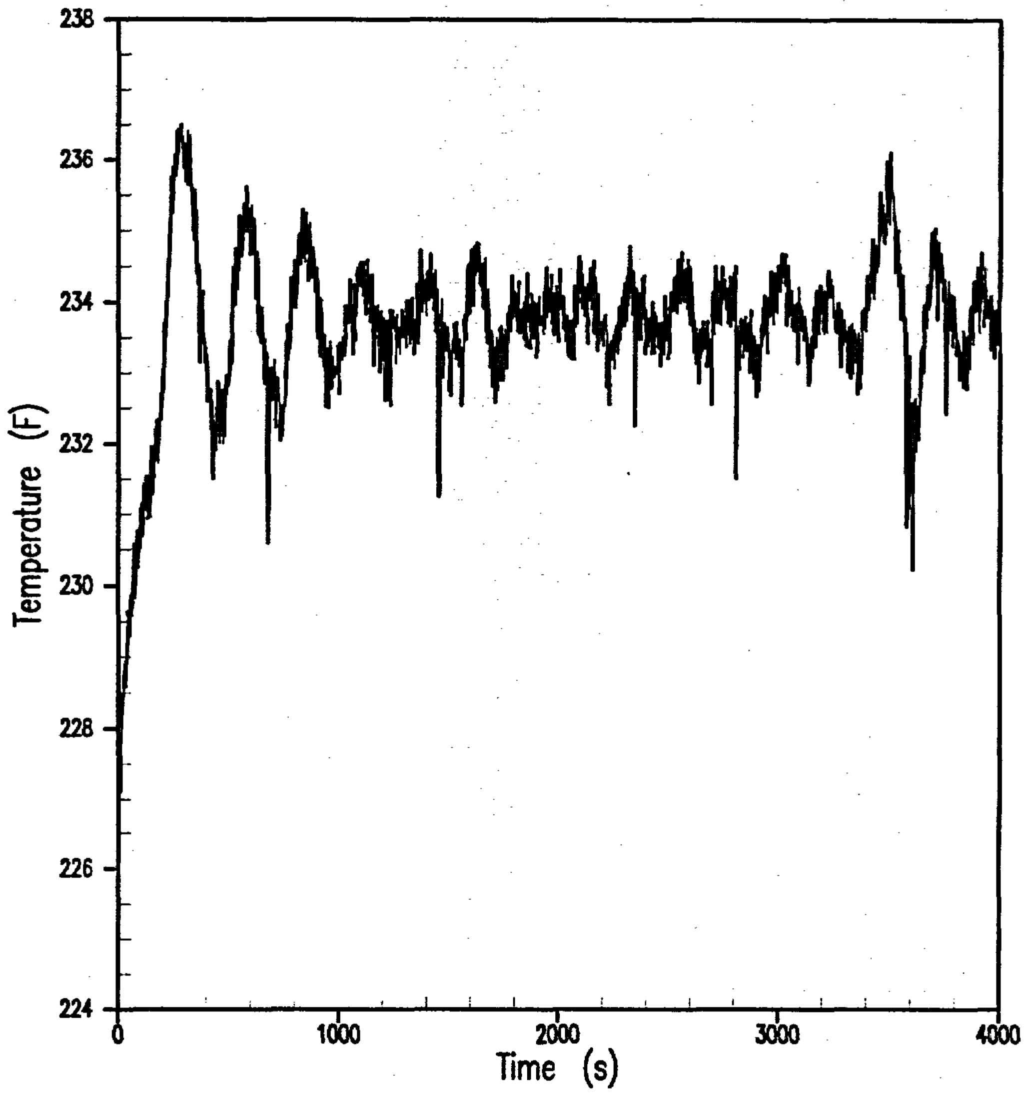

Figure 15.6.5.4C-26

~~PCT of the Hot Rod~~Hot Rod Cladding Temperature Near Top of Core

(Wall-to-Wall Floodup Case)

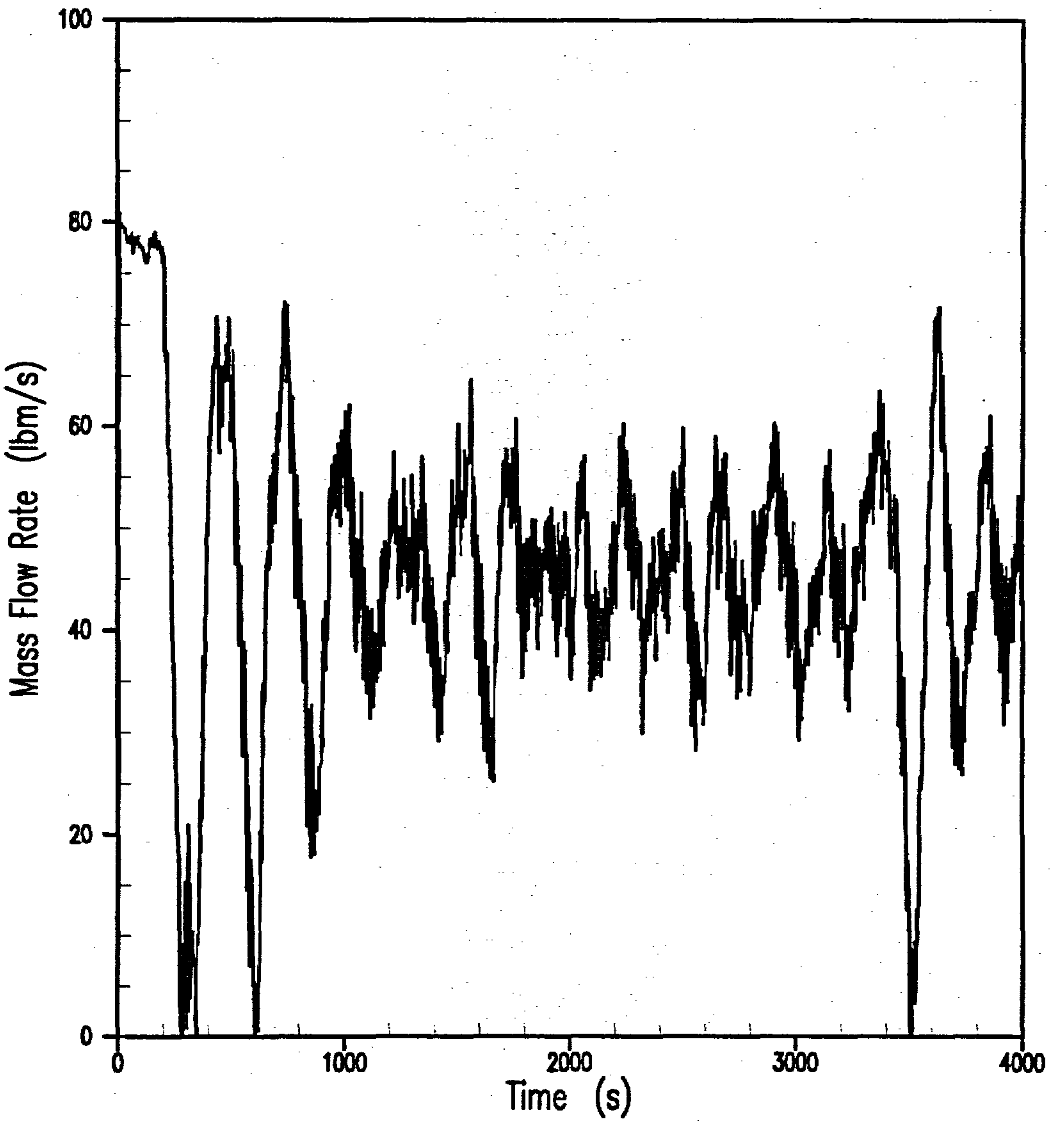

Figure 15.6.5.4C-27

DVI-A Mixture Flow Rate (Wall-to-Wall Floodup Case)

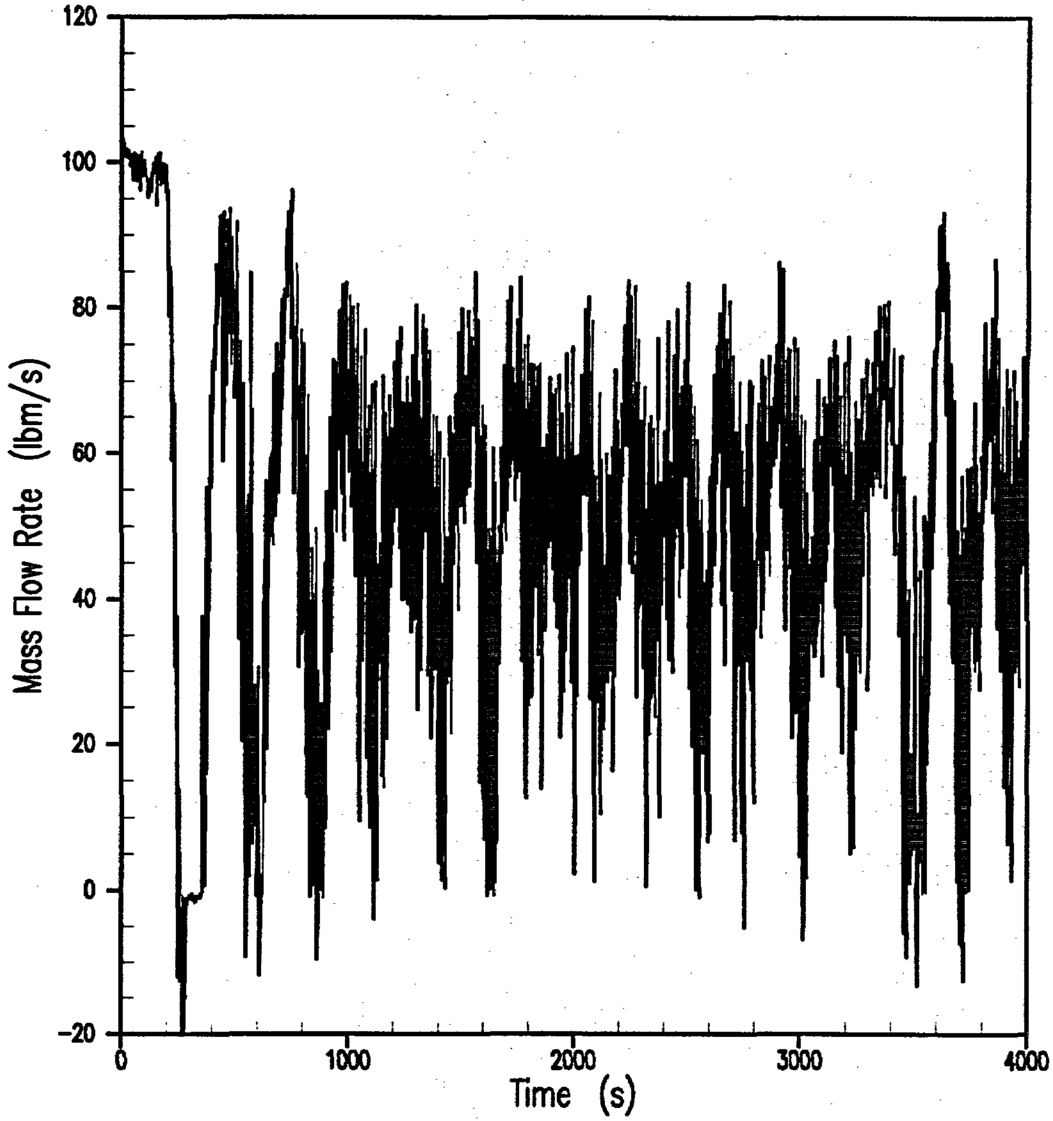

Figure 15.6.5.4C-28

DVI-B Mixture Flow Rate
(Wall-to-Wall Floodup Case)

PRA Revision:

None

WCAP Revision:

WCAP 15644 "AP1000 Code Applicability Report" Revision 1 includes the description and additional validation of the WCOBRA/TRAC long-term cooling model discussed in this response.