



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

U.S. Nuclear Regulatory Commission  
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/NRC1631

September 29, 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.

DO63

September 29, 2003

Correspondence with respect to the application for withholding should reference AW-03-1714, and should be addressed to Hank A. Sepp, Manager of Regulatory Compliance and Plant Licensing, 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  
Passive Plant Projects & Development  
AP600 & AP1000 Projects

**/Enclosure**

1. **Westinghouse Electric Company Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit AW-03-1714.**

**/Attachments**

1. **List of the AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses transmitted with letter DCP/NRC1631**
2. **Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated September 29, 2003**
3. **Non-Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated September 29, 2003**

DCP/NRC1631  
Docket No. 52-006

September 29, 2003

**Enclosure 1**

**Westinghouse Electric Company  
Application for Withholding and Affidavit**



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

September 29, 2003

AW-03-1714

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-03-1714 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-03-1714 and should be addressed to the undersigned.

Very truly yours,

A handwritten signature in black ink, appearing to read 'M. M. Corletti'.

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

/Enclosures

COMMONWEALTH OF PENNSYLVANIA:

SS

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared John S. Galembush, 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.

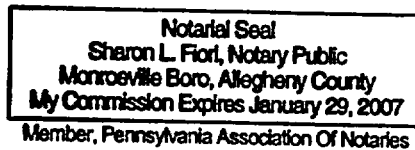


John S. Galembush, Acting Manager  
Regulatory Compliance & Plant Licensing

Sworn to and subscribed  
before me this 30<sup>th</sup> day  
of September, 2003



Notary Public



- (1) I am Acting Manager, Regulatory Compliance & Plant Licensing, of the Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company, LLC.
- (2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.
- (3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company, LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.
- (4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.
  - (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.
  - (ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

- (a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.
- (b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.
- (c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.
- (d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.
- (e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.
- (f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

- (a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.
- (b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

- (c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.
  - (d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.
  - (e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.
  - (f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.
- (iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.
  - (iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.
  - (v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in Attachment 2 as Proprietary Class 2 in the Westinghouse Electric Co., LLC document: (1) "AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Response."

This information is being transmitted by Westinghouse's letter and Application for Withholding Proprietary Information from Public Disclosure, being transmitted by Westinghouse Electric Company letter AW-03-1714 to the Document Control Desk, Attention: John Segala, CIPM/NRLPO, MS O-4D9A.



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

- (a) Provide documentation supporting determination of APP-GW-GL-700, "AP1000 Design Control Document," analysis on a plant specific basis**
- (b) Provide the applicable engineering evaluation which establishes the Tier 2 requirements as identified in APP-GW-GL-700.**

**Further this information has substantial commercial value as follows:**

- (a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for Licensing Documentation.**
- (b) Westinghouse can sell support and defense of AP1000 Design Certification.**

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

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

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

**Further the deponent sayeth not.**

## **PROPRIETARY INFORMATION NOTICE**

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

## **COPYRIGHT NOTICE**

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.790 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

September 29, 2003

**Attachment 1**

List of

Proprietary and Non-Proprietary Responses

<b>Table 1</b>	
<b>“List of Westinghouse’s Responses to DSER Open Items Transmitted in DCP/NRC1631”</b>	
*15.2.7-1P Item 1	21.5-1 Item 16
15.2.7-1 Item 1	21.5-1 Item 17
15.2.7-1 Item 2	*21.5-2P Item 18
*15.2.7-1P Item 3	21.5-2 Item 18
15.2.7-1 Item 3	*21.5-2P Item 19
15.2.7-1 Item 4	21.5-2 Item 19
15.2.7-1 Item 5	*21.5-2P Item 20
15.2.7-1 Item 6	21.5-2 Item 20
15.2.7-1 Item 7	*21.5-2P Item 21
15.2.7-1 Item 8	21.5-2 Item 21
15.2.7-1 Item 9	*21.5-2P Item 22
15.2.7-1 Item 10	21.5-2 Item 22
15.2.7-1 Item 13	21.5-2 Item 23
15.2.7-1 Item 14	21.5-2 Item 24
15.2.7-1 Item 15	21.5-2 Item 25
	21.5-2 Item 26
	21.5-2 Item 27
19.1.10.1-4 Rev 1	*21.5-2P Item 28
	21.5-2 Item 28
	*21.5-2P Item 29
	21.5-2 Item 29
	*21.5-3P Item 30
	21.5-3 Item 30

\*Proprietary

**Westinghouse Non-Proprietary Class 3**

**DCP/NRC1631  
Docket No. 52-006**

**September 29, 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: 15.2.7-1 Item 1

Original RAI Number(s): None

### *Summary of Issue:*

Westinghouse used selected G1 and G2 full-scale boil-off tests at pressure and power levels, which are prototypic of AP1000 conditions, to validate the WCOBRA/TRAC core model. The validation also determined, via sensitivity studies, that it was necessary to apply a corrective multiplier of 0.8 to the interfacial drag model to accurately predict the average core void fraction. However, Westinghouse stated that in the AP1000 DEDVI event during LTC, the average core exit quality is always less than 50%. This flow regime is quite different than the boil-off scenarios of the G1 and G2 tests. In the boil-off mode the exit quality is approximately 1.0.

Justify the applicability of the G1 and G2 tests to the AP1000 LTC conditions for use in validating the WCOBRA/TRAC results, and justify the validity of the corrective multiplier that was determined from benchmarks against the G1 and G2 tests for the AP1000 LTC model.

### **Westinghouse Response:**

The WCOBRA/TRAC validation presented in response to DSER Open Item 15.2.7-1 and discussed in the Revision 1 of WCAP-15644-P was extended by including the simulation of the post-quench phase for FLECHT-SEASET Run 31805. This is a 0.8 in/sec forced reflood experiment at 40 psia. During the test, the bundle is gradually quenched from bottom up. The experiment was continued for a time period following the bundle quenching time (~ 700 seconds) and useful data was collected during that period which was used to assess the WCOBRA/TRAC interfacial drag model. The conditions at 700 seconds are the following:

Pressure = 40 psia  
Average Linear Power = 0.22 kW/ft  
Inlet Flow = 0.81 in/sec  
Inlet Subcooling = 143 F

Once the bundle is quenched, a quasi-steady state is reached and the exit quality can be calculated from an energy balance by knowing the inlet flow and inlet subcooling. At 700 seconds the exit quality for this test was estimated to be equal to 50%. Therefore flow conditions at the top of the bundle are similar to the AP1000.

The FLECHT-SEASET test was simulated using WCOBRA/TRAC. The FLECHT-SEASET model is the same utilized for the validation of the Westinghouse WCOBRA/TRAC LBLOCA model and is discussed in WCAP-12945-P-A. The 12 ft heated section is divided in 14 axial nodes, which is consistent with G1 and G2 modeling guidelines discussed in the Revision 1 of WCAP-15644-P.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

The simulation was repeated using both the base interfacial drag model (YDRAG=1.0) and setting the multiplier YDRAG to 0.8 similar to the G1/G2 assessment.

The bundle is predicted to quench at about 650 seconds. Figure 1 shows the predicted collapsed liquid levels obtained with the base model (YDRAG=1.0). The two levels correspond to the inner and outer channels modeling the heated section. The levels are plotted in the time period which follows the reflood phase. After the bundle is quenched a quasi-steady state is reached where the collapsed liquid level slowly increases during the transient. The predicted collapsed liquid level is slightly higher in the periphery of the bundle than in the central region. On the average at 700 seconds, when the bundle is quenched, the collapsed liquid level is about 6.5 ft. This value was found to be in good agreement with the observation. The measured collapsed liquid level at 700 seconds is 6.44 ft.

The same calculation was repeated with YDRAG set to 0.8. Figure 2 shows that there is a small effect on the collapsed liquid level. Consistent with a reduction in interfacial drag the predicted collapsed liquid level is higher than the base case as more water is retained in the bundle, however the effect is small.

In conclusion, based on the results from the previous code validation against G1 and G2 experiments and additional results presented herein the WCOBRA/TRAC interfacial drag model is found to be adequate to model the AP1000 LTC conditions.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

### FLECHT-SEASET Run 31805 Base Model (YDRAG=1.0)

- Channel 2 (Inner Region)
- Channel 3 (Outer Region)
- ..... Data

a,b,c



Figure 1



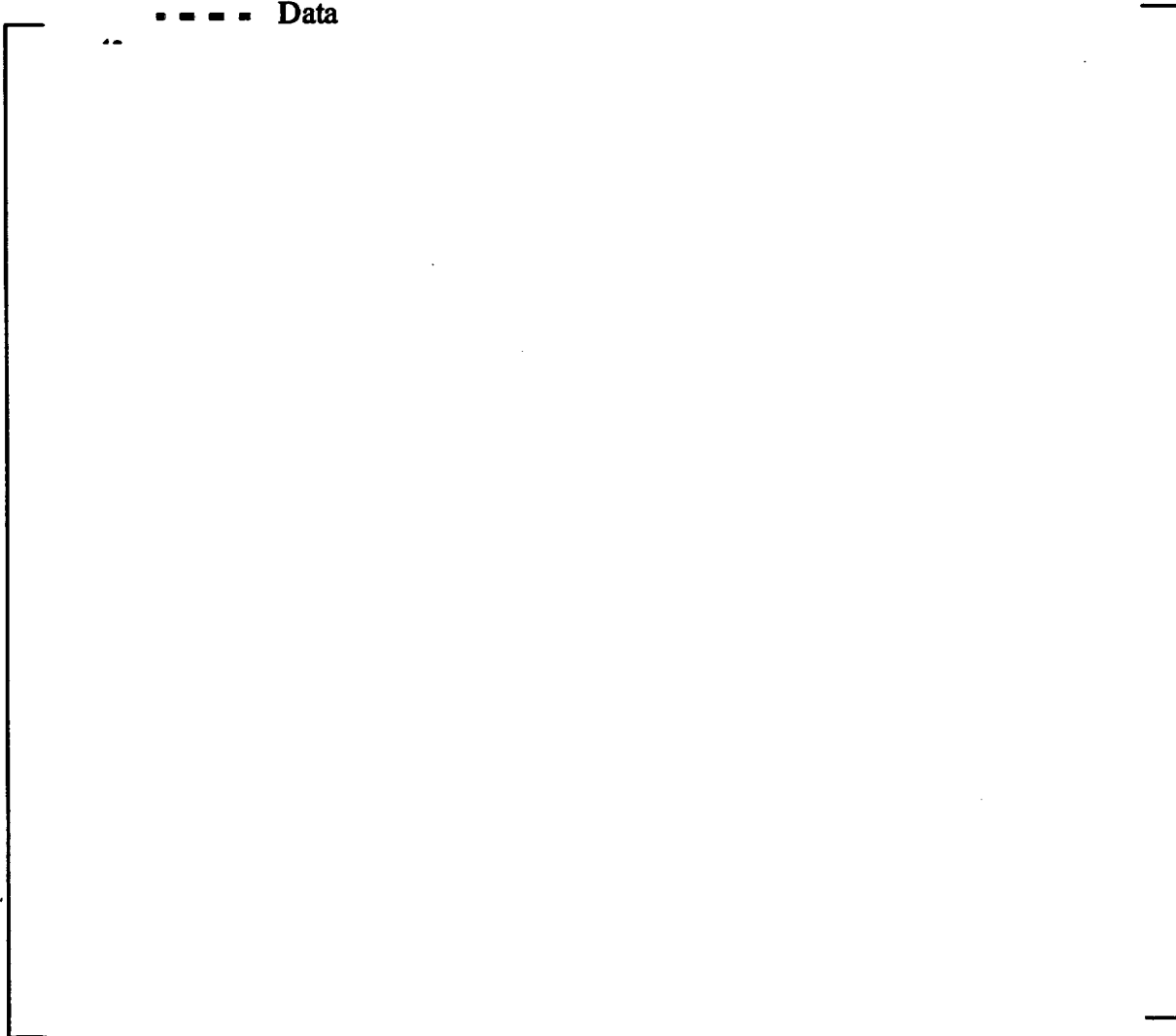
# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

### FLECHT-SEASET Run 31805 Reduced Drag (YDRAG=0.8)

- Channel 2 (Inner Region)
- Channel 3 (Outer Region)
- ..... Data



a,b,c

Figure 2

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**Design Control Document (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 Item 2

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

***Summary of Issue:***

DSER OI 15.2.7-1P Page 8 states that Figure 8 shows that the "mixture level" is located in proximity of the hot leg centerline, whereas Figure 8 shows collapsed liquid level in the hot leg.

Please clarify the discrepancy.

**Westinghouse Response:**

Figure 8 shows collapsed liquid level. The terminology "mixture level" in the text is incorrect and should be replaced with "hot leg collapsed liquid level".

**Design Control Document (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 Item 3

Original RAI Number(s): None

### Summary of Issue:

On DSER OI 15.2.7-1P Page 12, it is stated that "[the] expected flow regime at the top of the core is a churn or pulsating annular flow. The steam velocity is so low that entrainment of droplets is not expected to occur."

Indicate why this condition does not contradict the argument that sufficient liquid is entrained to avoid boron precipitation.

### Westinghouse Response:

As discussed in the DSER OI 15.2.7-1P Page 12, during the LTC, the vapour superficial velocity at the core exit is expected to be lower than 16 ft/s. This value was estimated based on the decay heat at 3000 seconds (velocity is based on the core cross section area). At that velocity the amount of liquid carried by steam main stream in the form of droplet (dispersed phase) is expected to be negligible.

However, on the average, liquid is still transported by the steam in the continuous field (film or slug) Note that the average core exit quality based on mass and energy balance is estimated to be in the 40% to 50% range as shown in the response to OI 15.2.7-1 item 11. Under those conditions a mixture level is expected to establish in the upper plenum. This was confirmed by WCOBRA/TRAC LTC calculations which indicated that liquid level reaches the hot legs. Because of flow area reduction from the core exit to the ADS4 lines, the vapour is subjected to a significant acceleration. At the beginning of the LTC (3000 seconds) the velocity at the core exit was estimated to be 16.0 ft/s. Based on that value, the velocity at different location was calculated based on the flow area reduction and shown in Table 1 and Figure 1.

The flow areas in Table 1 consider two how leg, two ADS4 line and three ADS4 valves.

**Table 1: Vapour Superficial Velocity at 3000 seconds**

a,b,c

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

a,b,c

Because of the high steam velocity, all liquid approaching the connection of the ADS4 line is expected to be entrained as shown in the schematic above. Note that if liquid is not entrained in the ADS4 stand pipe, the level in the hot leg will rise since core inlet flow always exceed the steam generation rate in the core, which is function of the decay heat. As the void fraction in the hot leg decreases, the liquid will be eventually pull through the tee by the high velocity steam. It

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

is also important to note that under these conditions the flow regime in the ADS4 line is highly dispersed and homogenous.

The following Tables 2 and 3 show the same calculation based on the LTC conditions corresponding to the 14 days case and 30 days. Despite the superficial vapour velocity at the core exit is below the flooding limit, the acceleration occurring in the hot leg and in the ADS4 lines is still significant. The Ku number calculated inside the ADS4 line is always greater than 3.2. The flow regime in the ADS4 line is expected to be annular-dispersed flow.

**Table 2: Vapour Superficial Velocity at 14 days**

a,b,c

**Table 3: Vapour Superficial Velocity at 30 days**

a,b,c

In conclusion the flow regime at the top of the core is quite different than the flow regime in the hot leg and ADS4 line. During the LTC the flow regime in the core is expected to be a churn or pulsating annular flow (chugging regime) whereas at the same time the flow regime in the ADS4 line is expected to be in annular dispersed flow. Therefore the conclusions presented on DSER OI 15.2.7-1P Page 12 do not contradict the argument that sufficient liquid is entrained to avoid boron precipitation.

**Design Control Document (DCD) Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number:** 15.2.7-1 Item 4

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

***Summary of Issue:***

In the revised DCD Section 15.6.5.4C.2 (DSER OI 15.2.7-1P Page 15), it is stated that there is a continuous flow of two-phase fluid into the hot legs, and mainly vapor flow toward the ADS Stage 4 valves occurs at the top of the pipe.

What is basis to conclude (at the end of the section) that the recirculation core liquid throughput is more than adequate to preclude any boron buildup on the fuel?

**Westinghouse Response:**

The statement regarding "mainly vapor flow toward ADS Stage 4 valves" is referring to the situation that the volumetric flow to the ADS Stage 4 valves is dominated by the vapor phase in this analysis. On a mass flow basis this flow has considerable liquid as can be observed by comparing the core vapor mass flow (Figure 15.6.5.4C-6) with the ADS Stage 4 mass flow rates (Figures 15.6.5.4C-9 and 10). It is this liquid flow through the core that leads to the conclusion that boron buildup due to boiling in the core will not result in boron precipitation on the fuel. This is discussed in more detail in the response to NRC 15.2.7-1 comment 11.

**Design Control Document (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 Item 5

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

***Summary of Issue:***

Since the DEDVI break long-term cooling case described in the revised DCD Section 15.6.5.4C) is the same case for the revised WCOBRA/TRAC LTC calculation, explain why the void fraction at the hot assembly top cell presented in Figure 11 (DSER OI 15.2.7-1P Page 11) and figure 15.6.5.4C-3 (DSER OI 15.2.7-1P Page 20) are different (other than the 2500 second shift).

**Westinghouse Response:**

Figure 11 has been generated by numerically smoothing the hot assembly top cell void fraction values of WCOBRA/TRAC, which are presented in Figure 15.6.5.4C-3. The smooth fit is accomplished using a locally weighted regression technique. This is the only figure in this response that was smoothed and it was done to see the void fraction value with the oscillations filtered out of the actual WCOBRA/TRAC calculation result.

**Design Control Document (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 Item 6

Original RAI Number(s): None

### *Summary of Issue:*

Tests at the AP1000 APEX facility indicate that the single failure of a stage 4-ADS valve produces lower core inventories if the failed valve is not in the pressurizer loop. The revised DCD Section 15.6.5.4C.2 (DSER OI 15.2.7-1P Page 14) states that the analysis of the DEDVI LTC case assumes failure of one of the two ADS-4 valves in the PRHR loop. Since the PRHR loop is the same loop that contains the pressurizer, justify that the small break analyses in Chapter 15 of the DCD are conservative in view of the test results from the APEX facility.

### **Westinghouse Response:**

To determine the impact of the assumed ADS Stage 4 valve failure location on the AP1000 plant, an ADS-4 failure location sensitivity study was performed with the NOTRUMP DEDVI simulation. For this simulation, the assumed ADS Stage 4 valve failure location was altered to the non-pressurizer loop (Defined as ADS 4-1 in the AP1000 plant model). The following describes the results obtained with this simulation.

As was observed in the AP1000 APEX facility, the ADS Stage 4 valve failure on the non-pressurizer side (ADS 4-1) resulted in a delay in the onset of IRWST injection flow (Figure 15.2.7-1). This can also be observed in the intact DVI line injection line flow as well (Figure 15.2.7-2). However, as observed in the AP1000 APEX simulations, the effect on the core/upper plenum two-phase mixture is minimal (Figure 15.2.7-3) with significant margin to core uncover being observed.

A comparison of the RCS inventory (Figure 15.2.7-5) and Vessel inventory (Figure 15.2.7-6) plots indicates that there is no significant change in the predicted minimum inventory observed for this simulation. What is observed is a shift in the time at which the minimum inventory is predicted to occur. A review of the ADS 1-3 behavior indicates no significant alterations in response (Figure 15.2.7-7 and Figure 15.2.7-8) for the ADS 1-3 liquid and vapor discharge paths respectively. As such, the difference in plant response is tied directly to the response of the ADS-4 flow paths.

A review of the total integrated ADS-4 discharge (Figure 15.2.7-9) indicates that approximately 100,000 lbm less discharge has occurred for the alternate ADS-4 valve failure location at transient termination than for the base model. This can be attributed directly to the change in IRWST injection performance. A review of the integrated IRWST injection flow (Figure 15.2.7-10) indicates that the added ADS-4 discharge in the base model can be attributed to excess IRWST injection.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

The nature of the AP1000 plant design is such that with the assumed failure of the ADS 4-1 path, significantly less liquid is discharged via this path (Figure 15.2.7-11) while the liquid discharge via the ADS 4-2 path increases only marginally (Figure 15.2.7-12). This is attributed to the interaction between the pressurizer and ADS 4-2 flow path.

With the assumed failure in the ADS 4-1 path, RCS depressurization is slowed (Figure 15.2.7-13) and IRWST injection is delayed (Figure 15.2.7-1); however, the vessel inventory is only minimally altered (Figure 15.2.7-6). A review of the pressurizer inventory (Figure 15.2.7-14) indicates that the initial decrease in ADS 4-1 discharge can be attributed to a combination of the failure assumption in the ADS 4-1 path and the mass storage differences in the pressurizer. This pressurizer mass storage behavior was also observed in the AP1000 APEX tests. The difference in pressurizer mass dominates until IRWST injection begins at which time excess inventory is discharged via the ADS 4 paths (Figure 15.2.7-9). No significant change in ADS-4 vapor discharge is observed between the two simulations (Figure 15.2.7-15).

A review of the vessel mass distributions indicates that the downcomer (Figure 15.2.7-16), core (Figure 15.2.7-17) and upper plenum (Figure 15.2.7-18) responses are virtually the same until the times at which IRWST injection begins to dominate. This indicates that the vessel is in a nearly quasi-equilibrium condition with a stable core average void fraction (Figure 15.2.7-19) and subsequently core collapsed coverage fraction (Figure 15.2.7-20). This indicates that the AP1000 plant design is such that as long as the vessel maintains an adequate source of inventory, core cooling can be assured.

For the long term cooling (LTC) analyses in Chapter 15, the earlier IRWST injection with the failure location on the pressurizer side results in less IRWST inventory at the beginning of the LTC analysis and earlier switchover to sump recirculation, as compared to the case with failure location on the non-pressurizer side. This is the case used in the Chapter 15 LTC analysis and is a more demanding situation for LTC analyses as compared to the non-pressurizer side failure location. As seen at the end of the NOTRUMP comparative plots (Figures 15.2.7-1 through 20) the quasi-steady conditions at 3000 seconds are reaching essentially the same conditions for the two failure locations.

These results indicate that the ADS-4 failure location does not significantly impact core cooling.

### Design Control Document (DCD) Revision:

None

### PRA Revision:

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Intact IRWST Injection Flow

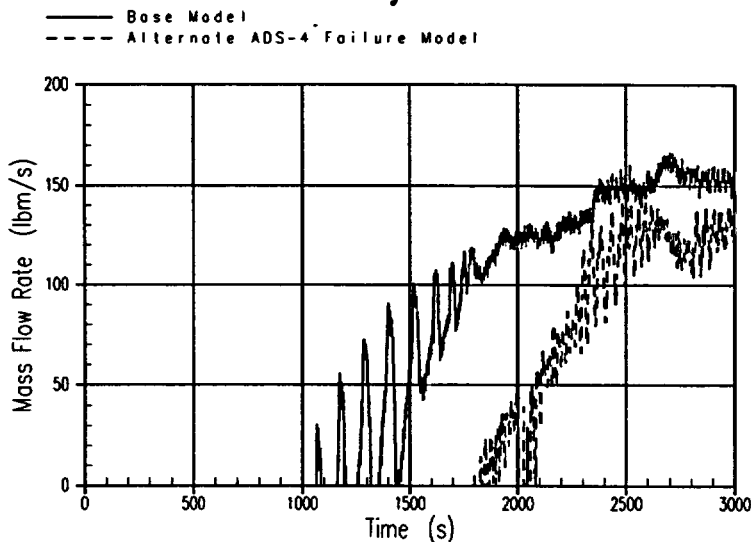


Figure 15.2.7-1 DEDVI: Intact IRWST Injection Flow

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Intact DVI Line Injection Flow

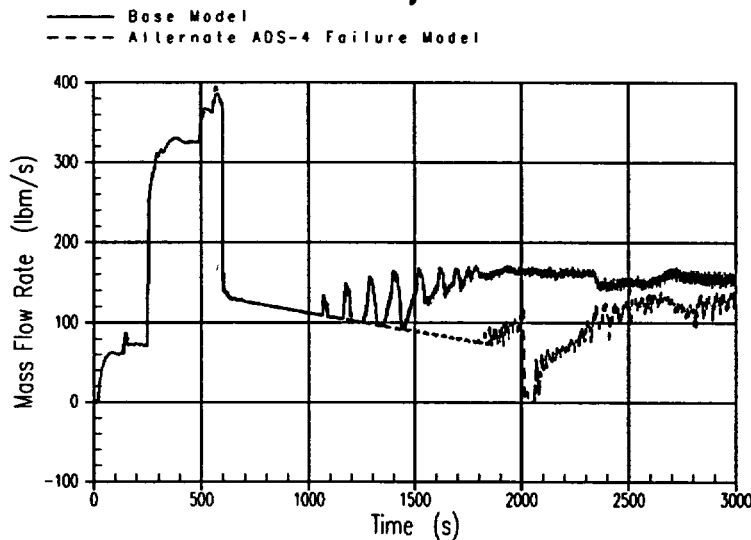


Figure 15.2.7-2 DEDVI: Intact DVI Line Injection

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Core/Upper Plenum Mixture Level

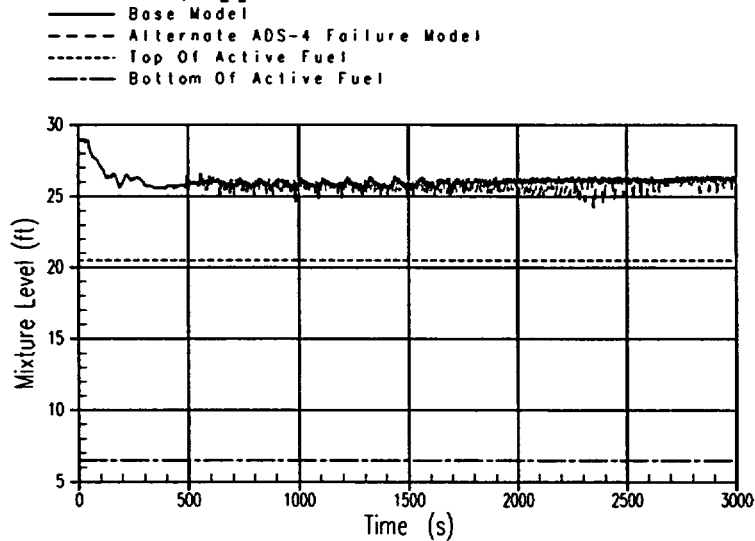


Figure 15.2.7-3 DEDVI: Core Upper Plenum Two-Phase Mixture Level

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Downcomer Mixture Level

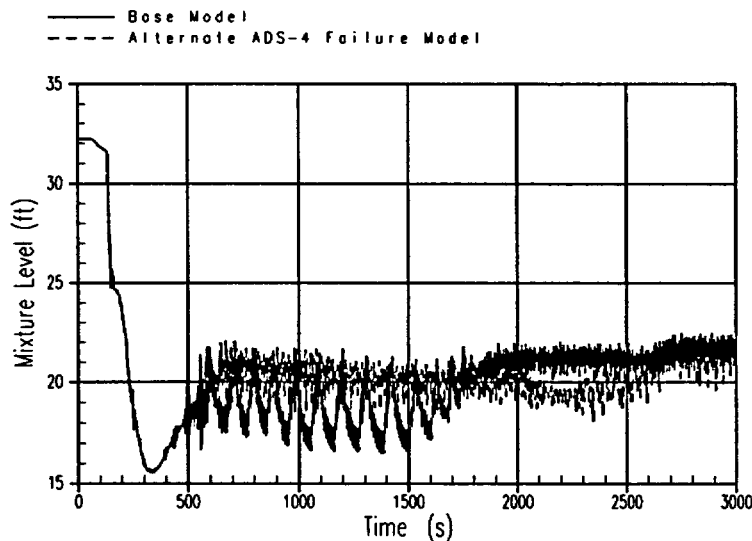


Figure 15.2.7-4 DEDVI: Downcomer Two-Phase Mixture Level

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results RCS System Inventory

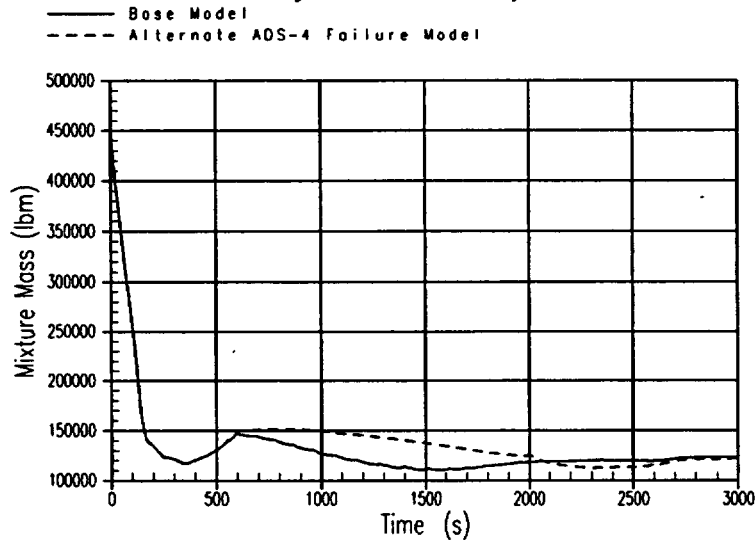


Figure 15.2.7-5 DEDVI: RCS Inventory

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Vessel Mixture Mass

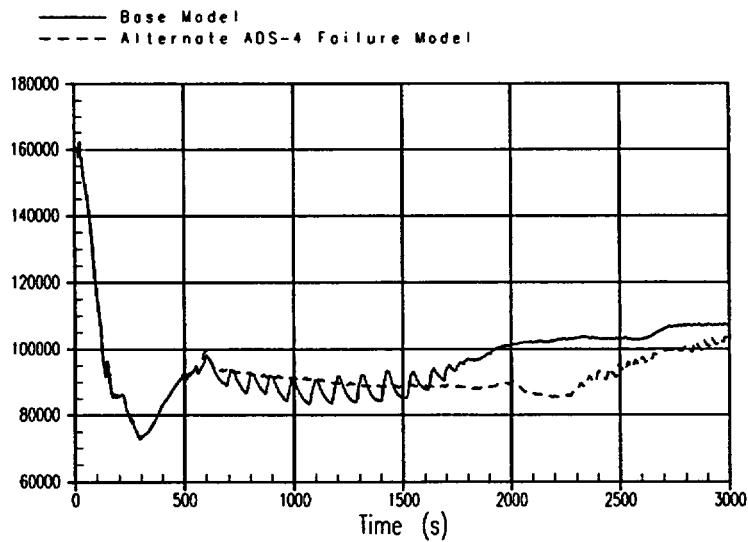


Figure 15.2.7-6 DEDVI: Vessel Mixture Mass

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results ADS 1-3 Liquid Discharge

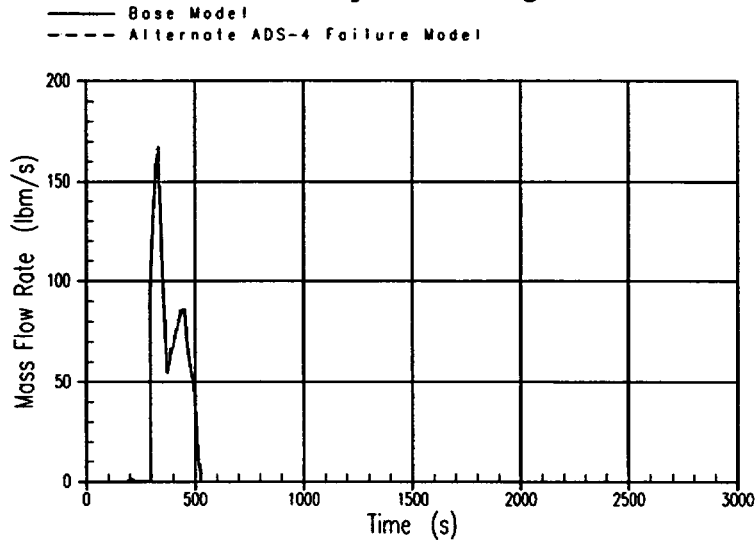


Figure 15.2.7-7 DEDVI: ADS 1-3 Liquid Discharge

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results ADS 1-3 Vapor Discharge

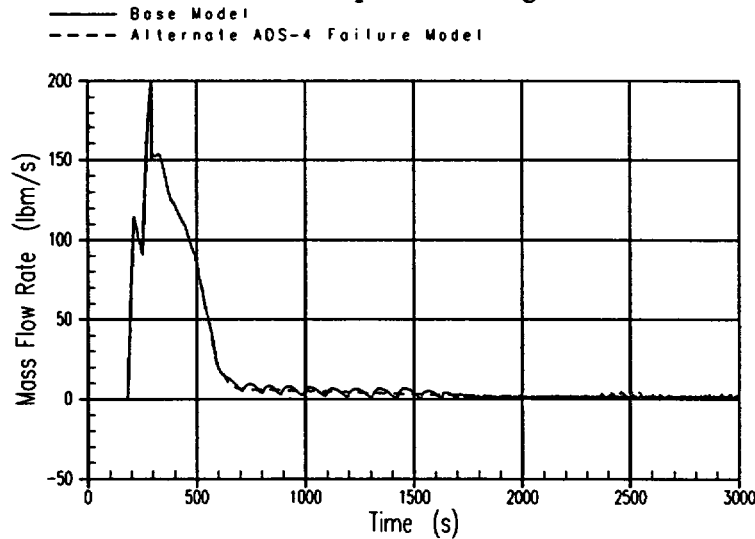


Figure 15.2.7-8 DEDVI: ADS 1-3 Vapor Discharge

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results ADS-4 Total Integrated Discharge

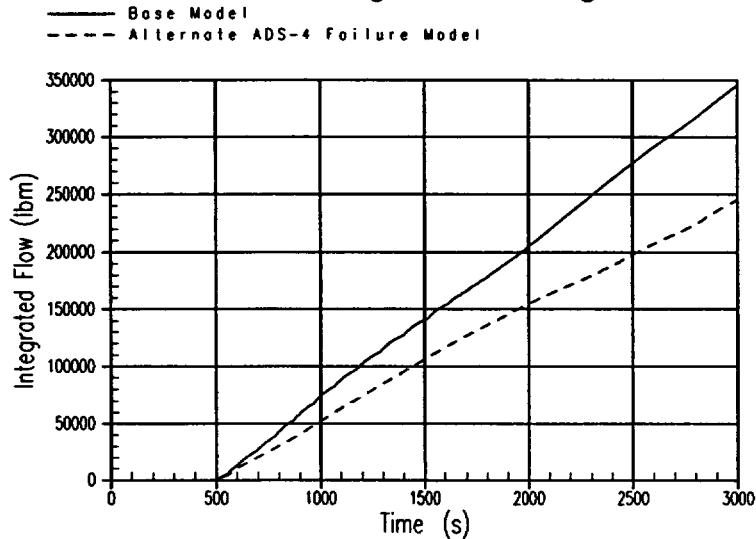


Figure 15.2.7-9 DEDVI: ADS 4 Total Integrated Discharge

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Intact IRWST Injection Flow

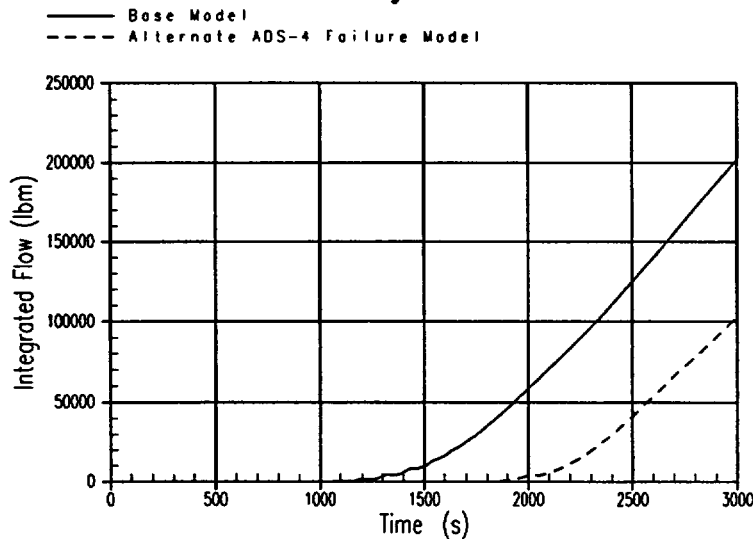


Figure 15.2.7-10 DEDVI: Integrated IRWST Injection Flow



# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results ADS 4-1 Integrated Liquid Discharge

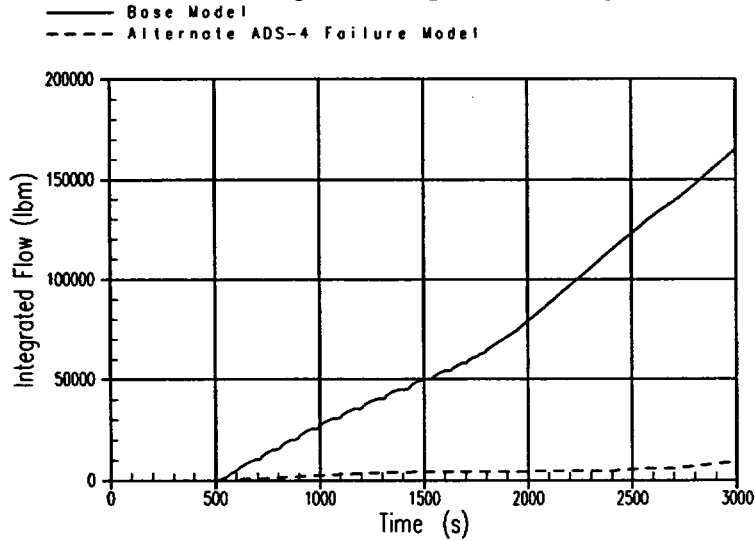


Figure 15.2.7-11 DEDVI: ADS 4-1 Integrated Liquid Discharge

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results ADS 4-2 Integrated Liquid Discharge

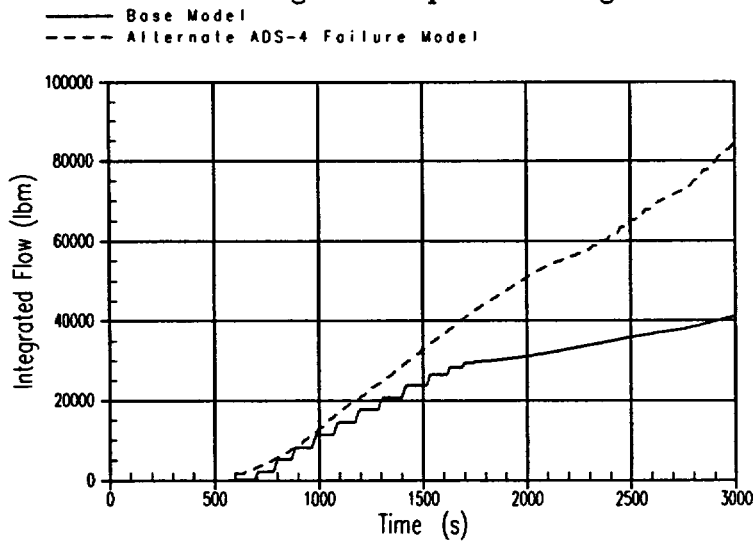


Figure 15.2.7-12 DEDVI: ADS 4-2 Integrated Liquid Discharge

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Downcomer Pressure At DVI Port

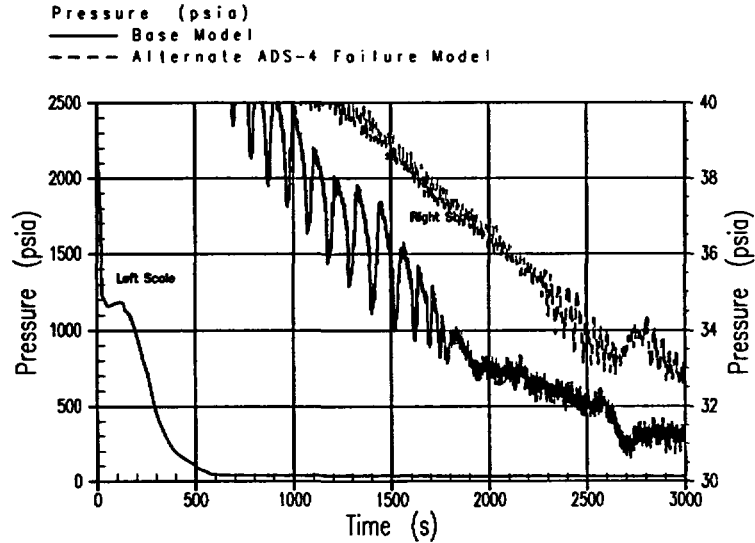


Figure 15.2.7-13 DEDVI: Downcomer Pressure At DVI Port

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Pressurizer Mixture Mass

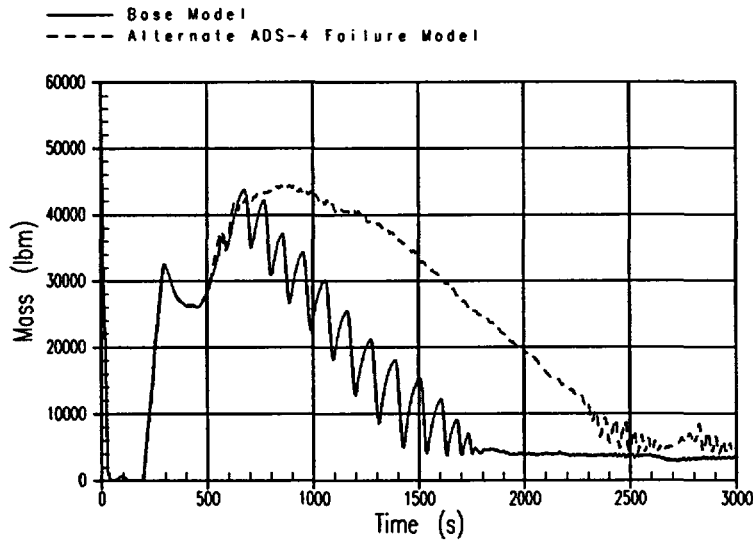


Figure 15.2.7-14 DEDVI: Pressurizer Mixture Mass

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results ADS-4 Integrated Vapor Discharge

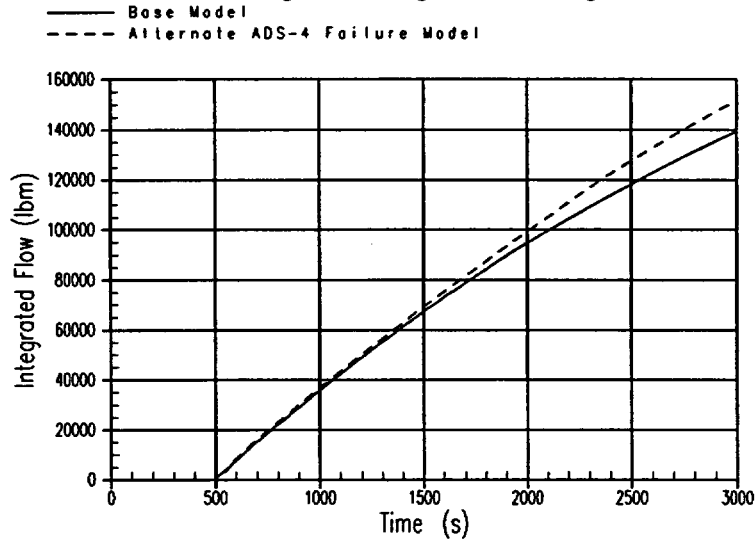


Figure 15.2.7-15 DEDVI: ADS 4 Integrated Vapor Discharge

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Downcomer Region Mixture Mass

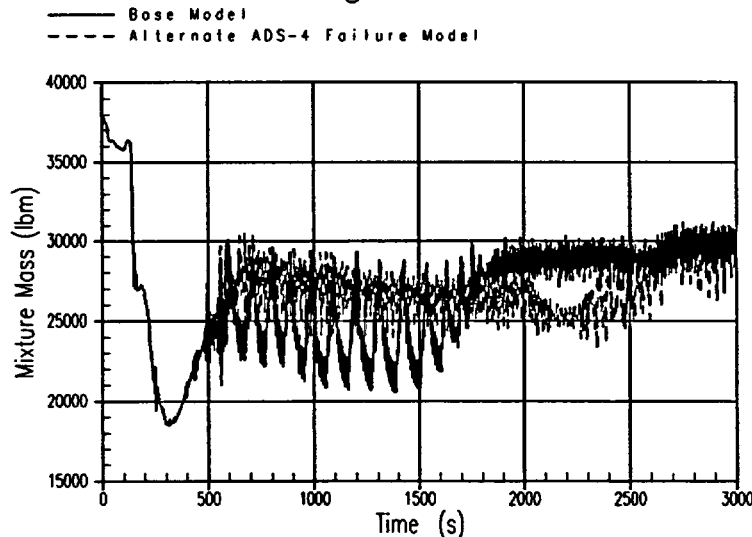


Figure 15.2.7-16 DEDVI: Downcomer Region Mixture Mass

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Core Region Mixture Mass

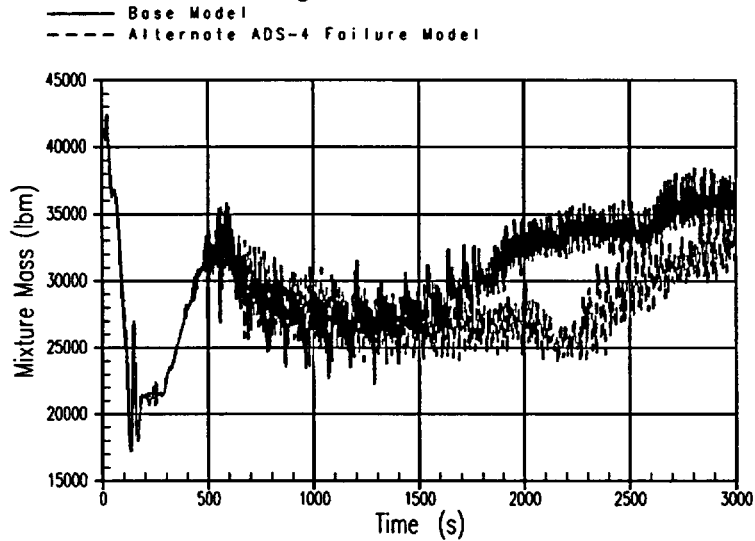


Figure 15.2.7-17 DEDVI: Core Region Mixture Mass

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Upper Plenum Mixture Mass

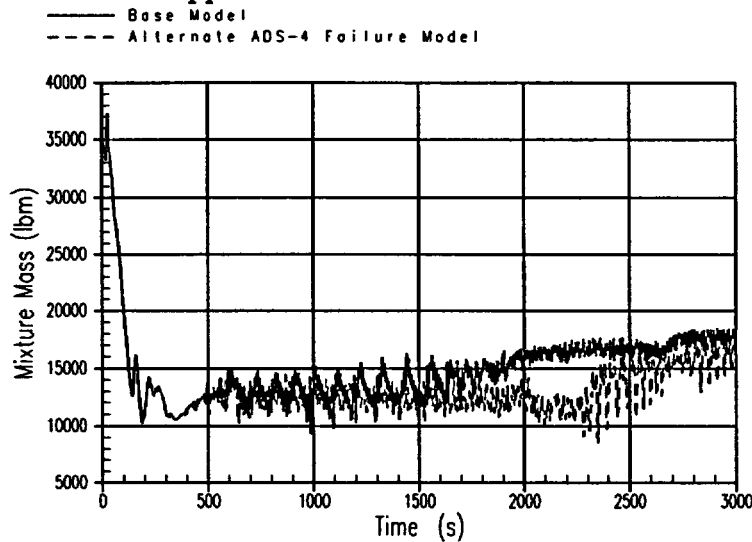


Figure 15.2.7-18 DEDVI: Upper Plenum Mixture Mass

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Core Region Average Void Fraction

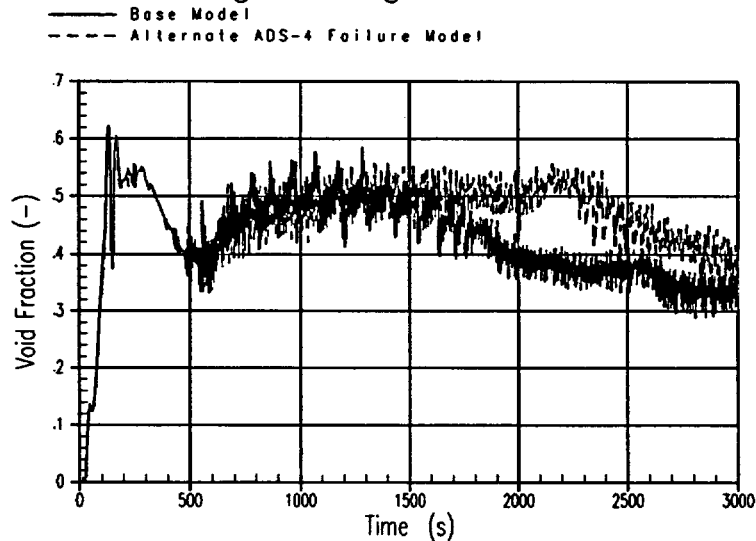


Figure 15.2.7-19 DEDVI: Core Average Void Fraction

### AP1000 NOTRUMP ADS-4 Failure Sensitivity Results Core Region Collapsed Level Percent

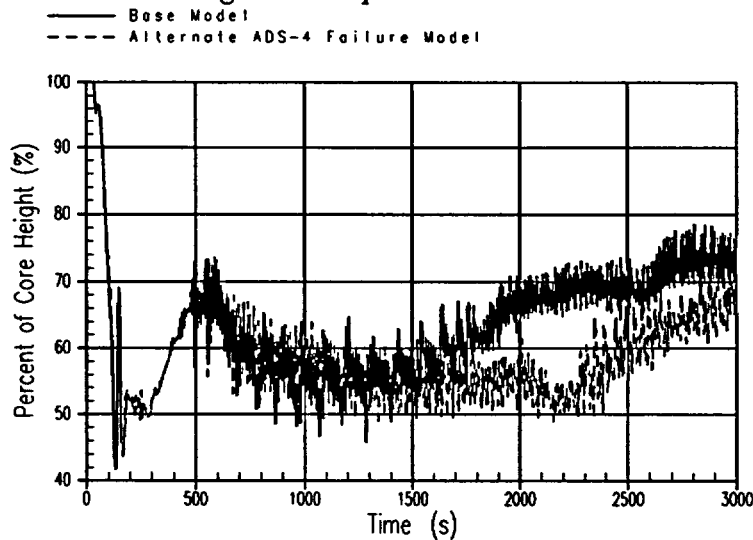


Figure 15.2.7-20 DEDVI: Percent Of Core Coverage

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number:** 15.2.7-1 Item 7

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

***Summary of Issue:***

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

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

**Westinghouse Response:**

The WGOETHIC model used to determine the containment pressure response for the DCD was used for the containment backpressure analysis with the following changes:

1. The DCD model is biased to maximize containment pressure. These assumptions were changed for the backpressure analysis to minimize containment pressure.
  - Heat transfer coefficients multipliers which are set to values less than unity for the peak pressure analysis are set to unity for the backpressure analysis
  - Heat sinks that are conservatively neglected for the peak pressure analysis are included for the backpressure analysis
  - Initial conditions inside containment that are biased to the highest operating pressure and temperature are set to the lowest operating pressure and temperature. Relative humidity is set to 100% to minimize the initial air inventory inside containment. Environmental boundary conditions are biased to maximize heat transfer from the passive containment cooling system and minimize the containment pressure.
  - The containment vent system is assumed to be open at the start of the event and closes on an SI signal. This allows an initial decrease in the air inventory which results in a lower containment pressure
2. Mass and energy release rates that are specific for the double-ended DVI break are included in the WGOETHIC model.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

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

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

The containment pressure for this event is shown in Figure 1. Also shown are the times for ADS-4 actuation, IRWST actuation and switch over to sump recirculation. This figure shows that 25 psia is an acceptable backpressure for the DEDVI LOCA analysis for the time period following the initial blowdown until the initiation of sump injection. For the first 200 seconds, the pressure is below 25 psia, but during this time, the RCS pressure is high and the containment backpressure has little effect on the plant response. At the time of IRWST injection, the backpressure is well above 25 psia. As the IRWST drains, the pressure continues to fall reaching approximately 24.5 psia at sump recirculation. Thus, for the short-term analysis, the backpressure of 25 psia is a conservative limit. A lower pressure boundary condition is appropriate for times after sump recirculation is established.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 Containment Backpressure for DEDVI

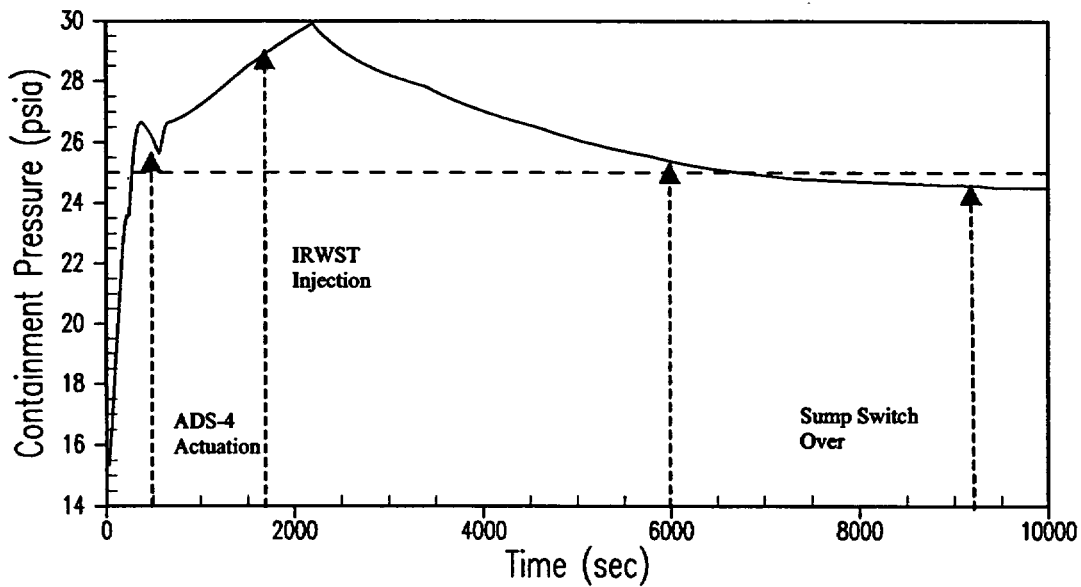


Figure 1: Containment Backpressure for DEDVI

Design Control Document (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 Item 8**

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

***Summary of Issue:***

The revised DCD Section 15.6.5.4C.1 (DSER OI 15.2.7-1P Page 13) states that AReference 24 [WCAP-15644, "AP1000 Code Applicability Report."] provides details of the AP1000 WCOBRA/TRAC modeling. The coarse reactor vessel modeling used for AP600 has been replaced with a detailed nodding like that applied in the large-break LOCA analysis described in subsection 15.6.5.4A. Also, in the revised DCD Section 15.6.5.4C.3 (Westinghouse letter DCP/NRC1617, dated September 8, 2003), it is stated that in WCOBRA/TRAC analysis, the core is nodalized as described in Reference 24. However, neither WCAP-15644 nor DCD Subsection 15.6.5.4A provides detailed AP1000 WCOBRA/TRAC modeling.

- A. Please clarify where the core nodalization is described for the LTC analysis.
- B. Clarify whether the core nodalization is the same as that described in your ASummary@ of the response to Open Item 15.2.7-1, which states that for the AP1000 LTC model, the core region was subdivided axially into 17 nodes.

**Westinghouse Response:**

- A. The revised AP1000 WCOBRA/TRAC modeling is described in Section 2.3.3 of WCAP-15644-P Revision 1 that was submitted by Westinghouse letter DCP/NRC1627 dated September 19, 2003.
- B. The revised AP1000 WCOBRA/TRAC model divides the active core region into 17 axial nodes.

**Design Control Document (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 Item 9

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

***Summary of Issue:***

DSER OI 15.2.7-1P Page 32 indicates WCAP-15644 will be revised to include the description and additional validation of the WCOBRA/TRAC LTC model discussed in this response. This is a confirmatory item.

**Westinghouse Response:**

Westinghouse and the NRC held a teleconference to discuss this open item. The additional validation requested is discussed in the Revision 1 of WCAP-15644 that was transmitted to the NRC in Westinghouse letter DCP/NRC1627 dated 9/19/2003.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

DSEI Open Item Number: 15.2.7-1 Item 10

Original RAI Number(s): None

### *Summary of Issue:*

The NRC staff is attempting to modify the RELAP5 AP1000 model to evaluate LTC. The current model does not model sump recirculation into the DVI lines. Please provide the following information to enable the staff to perform LTC confirmatory analyses with RELAP5 for a postulated double DVI line break.

- A. Containment temperature, pressure, water level and boric acid concentration versus time for 30 days.
- B. Recirculation line lengths, areas, elevations and resistances. All elevations including containment water levels should be given relative to the reactor vessel DVI nozzle elevation.
- C. Recirculation valve actuation setpoints and the WCOBRA/TRAC calculated time for sump recirculation following a double ended DVI line break.
- D. Sump screen resistance as a function of flow blockage.

### **Westinghouse Response:**

In a conference call between NRC and Westinghouse on September 16, 2003, it was agreed that Westinghouse would perform LTC sensitivity analyses using the WCOBRA/TRAC LTC model to address NRC interests rather than NRC creating a RELAP5 LTC model. NRC took the action to specify in the near term the sensitivity analyses of interest.

### **Design Control Document (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 Item 13

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

***Summary of Issue:***

Please provide justification for the mixing volume assumed in the boric acid precipitation analysis.

**Westinghouse Response:**

The core region volume is minimized in order to maximize the rate of core boron concentration increase. The regions that are assumed to mix include the core itself and the upper plenum / HLs. The upper plenum and HLs are included because the detailed long-term cooling WCOBRA-TRAC analysis shows that there is recirculation of water from the upper plenum and HLs back into the core. The amount of water assumed in the boron concentration analysis was taken from this LTC WCOBRA-TRAC analysis.

Note that using a smaller mass of water would have no impact on the peak core boron concentration. This is because the initial bounding ADS 4 vent quality of 60% continues on long enough that the core boron concentration reaches a steady state equilibrium value. The maximum core concentration is reached in about 2.4 hours, although the concentration gets very close to that value much faster (~ 0.5 hr). The 60% ADS 4 vent quality continues on for more than 3 hours. As a result, decreasing the core region mixing volume will not impact on the peak boron concentration

**Design Control Document (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 Item 14**

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

***Summary of Issue:***

Does the head of water in the core and upper plenum consider the additional mass due to the boric acid that concentrates in this region? If not, please explain the omission, especially when sensitivity studies show high concentrations in the core. How does the 35,000ppm concentration in the core, lower plenum, and upper plenum and other portions of the hot side of the RCS affect the flow rate entering the inner vessel? Please explain.

**Westinghouse Response:**

The specific gravity of boric acid becomes slightly greater than 1 at boron concentrations above 10,000 ppm. So with the peak boron concentration calculated for AP1000 (7400 ppm) the T/H analysis would not be affected.

Even if a boron concentration as high as 35,000 ppm is arbitrarily assumed, the specific gravity is only slightly higher than water (about 1.02). Such high core boron concentrations would require very high ADS 4 vent qualities. With the small increase in specific gravity and the smaller amount of water in the reactor core, upper plenum/HL and ADS 4 (due to the high ADS 4 vent quality), the T/H analysis would not be significantly affected.

**Design Control Document (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 Item 15

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

***Summary of Issue:***

It is stated that the analysis of the boron concentration in the core (Attachment 1: AP1000 Long Term Boron Concentration Evaluation, Section 3) assumed the core to only mix with the water in the upper plenum having a minimum mass of 27,490 lb, which is based on WCOBRA-TRAC analysis.

What is the basis of the minimum mass of 27,490 lb in the precipitation analysis? Please explain.

**Westinghouse Response:**

The basis for the minimum mass is the detailed long-term cooling WCOBRA-TRAC LTC analysis. Note that as discussed in 15.2.7-1 Item 13, this mass has no impact on the peak core boron concentration in the AP1000.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number:** 19.1.10.1-4 Revision 1

**Original RAI Number(s):** 720.027

**Summary of Issue:** Impact of Uncertainties on PRA Results and Conclusions

The staff review has identified two AP1000-specific areas of uncertainty which individually, or collectively with other areas of uncertainty (e.g., uncertainty associated with failure probabilities of squib valves), have the potential to affect the PRA results and conclusions regarding the need for "certification requirements," such as ITAACs, RTNSS and COL action items. These uncertainties have also the potential to increase the number of "low margin risk significant" sequences which should be analyzed conservatively to bound thermal-hydraulic (T-H) uncertainty and determine success criteria for systems and operator actions. These two areas of uncertainty are discussed below.

One area of uncertainty is related to initiating event frequencies assumed in the PRA. The staff requested additional information (see RAI 720.027) about differences in initiating event category frequencies used in the AP600 and the AP1000 PRAs for large LOCAs and SGTR accidents. The applicant's response to RAI 720.027 did not address adequately the basis for the decrease of such initiating event frequencies which have a significant impact on the PRA results. For the large LOCA category, the applicant states that in the AP1000 PRA "operating experience" data reported in NUREG/CR-5750 for pipe breaks were used. However, the NUREG/CR-5750 data rely on expert opinion and include significant uncertainty. In addition, since NUREG/CR-5750 was published additional information (e.g., Davis Besse finding) is available. For SGTR events, the frequency used in the AP1000 PRA is based on a more recent calculation that was performed in conjunction with a replacement steam generator project which is proprietary to the applicant. The staff believes that the impact of uncertainties on PRA results and insights, associated with the frequencies of large LOCAs and SGTR accidents assumed in the AP1000 PRA, needs to be investigated and addressed appropriately by the design certification process.

A second area of uncertainty is related to the success criteria assumed in the AP1000 PRA for passive containment cooling by air flow. The AP1000 PRA event trees include a top event for containment cooling (event CHR). It is stated that *"For success paths that result in steam release to the containment, the success of containment cooling (PCS or RNS) is modeled. If containment cooling is successful, then the path ends in an OK state. If PCS water cooling is not successful, then the path goes to a special OK end state to allow containment integrity sensitivity studies to be made."* This "special OK" end state is labeled "late containment failure (LCF)" end state and defined as an end state *"...where the containment heat removal by either passive containment cooling system (PCS) or component cooling water (CCS) heat exchangers via normal residual heat removal (RHR) fails."* The staff requested clarification (see RAI 720.030) about the meaning of the "special OK" status. The applicant responded that a sensitivity study shows that even if the LCF state is considered to be a core damage, the plant CDF would increase by only 29 percent. The staff needs further information regarding the impact of this assumption on the focused PRA, where no credit is taken for the non-safety-related systems, and on the RTNSS process.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

The impact of these two areas of uncertainty on the results of the PRA, including the PRA results used in the RTNSS process, should be addressed in the design certification process. This is Open Item 19.1.10.1-4.

### Westinghouse Response: (Revision 1)

Westinghouse revised its response to RAI 720.027 in order to address the NRC concerns about the initiating event frequency of large LOCAs. The response to RAI 720.030 was also revised in order to address the NRC concerns about the initiating event frequency of steam generator tube ruptures. These revised responses (rev. 1) were submitted to the NRC on March 25, 2003 in letter DCP/NRC1556. Westinghouse also revised the discussion on uncertainties in the AP1000 RTNSS evaluation (WCAP-15985, rev. 1) to reflect these RAI responses as well as other uncertainties. This revision was included in a revised response to RAI 720.039 (rev. 2). This revised response was submitted to the NRC on April 4, 2003 in letter DCP/NRC1565.

### LCF End State

At the beginning of the PRA process, it was not known whether air cooling of the passive containment would be sufficient to prevent containment failure. It was postulated that accident sequences with successful emergency core cooling and PCS water failure could potentially produce core damage if the containment failed. The core damage was postulated to be initiated in the long-term due to loss of water from the system by steaming or draining from a failed containment. The LCF end state was created so that these potential long-term large release sequences could be tracked and treated appropriately as the sequences became more understood. LCF was treated as a sensitivity case, and the frequency of the accident class is approximately 25% of the base case core damage frequency.

During the PRA, an analysis of the containment capability during air cooling was performed and is documented in Chapter 40 of the PRA. The conclusion of the analysis is that the nominal long-term conditional containment failure probability for the containment with air cooling alone is 2.2%. Therefore, it can be concluded that there is only a small probability that sequences in the LCF accident class would fail the containment. Thus, only a small fraction of the LCF frequency could contribute to large release.

If containment failure were assumed to occur in an LCF accident sequence, the progression of the accident to a large early release is unlikely. Although failure of the containment precedes core damage in the accident sequence, core damage would not be expected to begin for many hours after containment failure. The most likely progression to a severe accident is via water loss through steaming through the break. The rate of the decay heat steaming after being shut down for almost 2 days is very small. It would take many hours to lose sufficient water from the containment to begin melting the core. Containment failure would not be a difficult event to diagnose for the operator.



# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

Therefore, accident management strategies could be implemented to prevent core damage, even on an ad hoc basis.

Thus, it is concluded that LCF sequences do not contribute significantly to core damage or large early release frequency. The probability that the containment will fail is small, and even if the containment fails, the accident will not likely proceed to a large early release with no time for accident management to prevent core damage or mitigate the consequences of a release. Therefore, there is no additional treatment needed for LCF sequences in the PRA or in RTNSS.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

DSER Open Item Number: 21.5-1 Item 16

Original RAI Number(s): None

### *Summary of Issue:*

For the comparison of the integrated liquid discharge through ADS-4 between the base DCD case and the revised nodding model, Figure 21.5-1.12 on DSER OI 21.5-1P Page 11 has an inconsistency in that the heading states "ADS-4 Liquid Discharge Comparison" whereas the actual figure is for ADS-4 integrated vapor discharge.

Please clarify.

### **Westinghouse Response:**

Figure 21.2.1-12 was intended to be a plot of ADS-4 Liquid Discharge Comparison. However the figure that was included was ADS-4 integrated vapor discharge as the Item states. The attached figure is the corrected Figure 21.5-1.12 that shows ADS-4 Liquid Discharge Comparison.

### **Design Control Document (DCD) Revision:**

None

### **PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP Entrainment Study Results ADS-4 Integrated Liquid Discharge

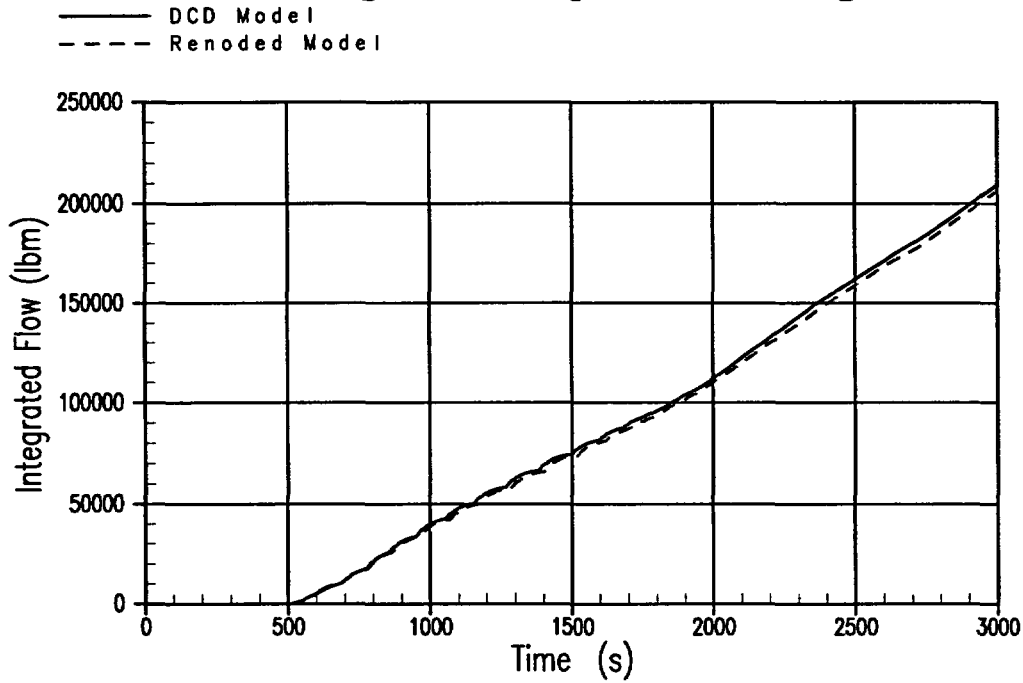


Figure 21.5-1.12 ADS-4 Liquid Discharge Comparison

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number: 21.5-1 Item 17**

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

***Summary of Issue:***

The text on DSER OI 21.5-1P Page 3 states that Figure 21.5-1.14 presents a comparison of the upper downcomer pressure between the base case and the sensitivity case, but the actual figure is for pressurizer pressure (except for the heading). The text also states that the pressurizer mixture level response (Figure 21.5-1.25) reflects the change in pressure response (Figure 21.5-1.14) observed in the model.

Clarify the discrepancy regarding Figure 21.5-1.14.

**Westinghouse Response:**

Figure 21.5-1.14 was intended to show the upper downcomer pressure between the base case and the sensitivity case, but the actual figure is incorrect as stated in Item 17 above. The attached figure provides the corrected Figure 21.5-1.14. The plot of Pressurizer pressure is also provided as 21.5-1.26 and the text should refer to this figure for pressurizer pressure. No update to the previous response is intended.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP Entrainment Study Results Downcomer Pressure At DVI Port

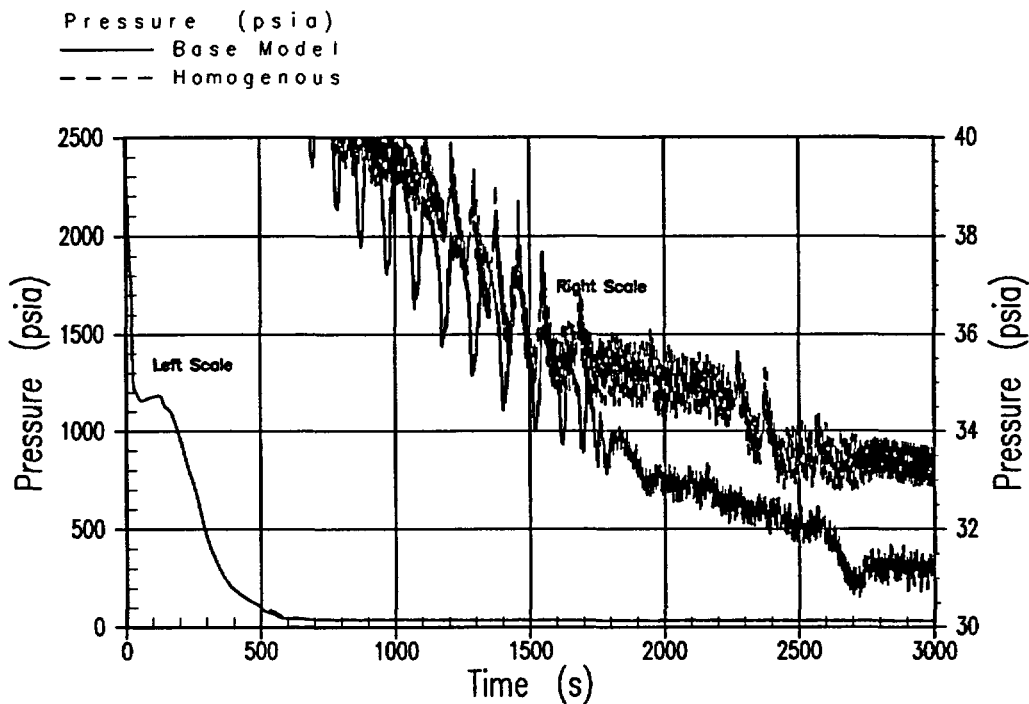


Figure 21.5-1.14 Downcomer Pressure Comparison

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 NOTRUMP Entrainment Study Results Pressurizer Pressure

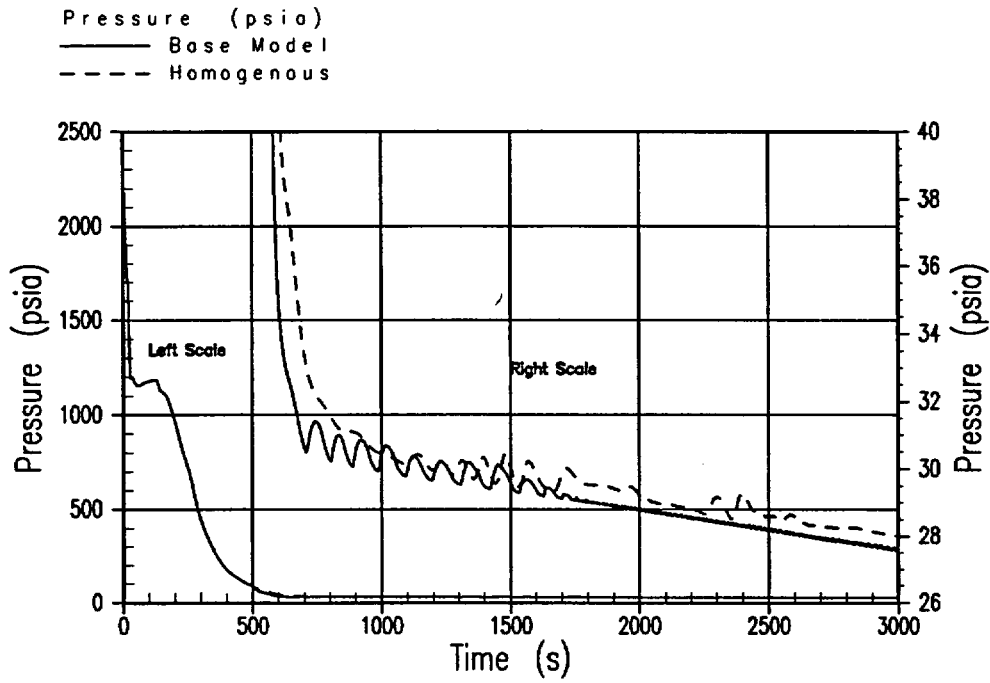


Figure 21.5-1.26 Pressurizer Pressure Comparison

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number: 21.5-2 Item 18**

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

### ***Summary of Issue:***

In the response to DSER Open Item 21.5-2P (Westinghouse letter DCP/NRC1611, August 13, 2003), the core collapsed liquid level for AP1000 APEX tests 02 and 03 are given in Figures 7, 11, 21.5-2.17, 21.5-2.19, and 21.5-2.44. Report OSU-APEX-03002 "OSU APEX-1000 Test Facility Description Report" indicates that uncompensated core level is provided by instrument LDP-118. The core collapsed liquid level figures for the tests performed are provided in the CDs attached to the Test Acceptance Reports OSU-AP1000-02 through OSU-AP1000-05.

- A. Why do the test acceptance reports not present the core collapsed liquid level figures for various tests? Please incorporate these figures in the test acceptance reports.
- B. Provide and justify the modifications that are made to the uncompensated core level to produce the results presented in the response to DSER OI 21.5-2.

### **Westinghouse Response:**

- A. The core collapsed level is included in Appendix B of the report. They are also included in Figures 1-5 for the five matrix tests completed.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



**Figure 1: Core Collapsed Level for Test DBA-01**



# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



**Figure 2: Core Collapsed Level for Test DBA-02**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



**Figure 3: Core Collapsed Level for Test DBA-03**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



**Figure 4: Core Collapsed Level for Test DBA-04**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



**Figure 5: Core Collapsed Level for Test TR-02**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

- B. The OSU data analysis program adjusts the collapsed level that is measured by the LDP instruments by:

$$CLDP = LDP * \rho_{ref} / \rho_{local}$$

Where:

CLDP is the compensated collapsed liquid level

LDP is the uncompensated collapsed liquid level as measured in the test

$\rho_{ref}$  is the liquid density in the reference leg of the LDP cell

$\rho_{local}$  is the local liquid density at the temperature and pressure at the LDP cell location

### Design Control Document (DCD) Revision:

None

### PRA Revision:

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

DSER Open Item Number: 21.5-2 Item 19

Original RAI Number(s): None

### *Summary of Issue:*

As mentioned in the ACRS Meeting in Monroeville in July, 2003, the APEX test facility contains an oversized downcomer. The oversized downcomer will produce high liquid inventories for extended periods of time which will maximize the liquid and two-phase levels in the core and upper plenum. This suggests the APEX facility cannot be used to simulate the minimum liquid and two-phase levels in the inner vessel that could occur following small breaks in the AP1000 plant. With a larger downcomer, more liquid mass will be retained in the vessel for small breaks. The statements in the Westinghouse August 13, 2003 letter (DCP/NRC1611) that the APEX-1000 facility is well scaled to AP1000 and the two-phase level remains in the upper plenum while the core remains covered for all phases of the simulated accident may not be appropriate and is misleading.

Please discuss the impact of the larger downcomer on the relevant APEX tests and explain why the facility test results can be used to demonstrate that significant amounts of inventory in this facility apply to the anticipated AP1000 response. Please also explain the statement that the APEX tests show the insensitivity of the AP1000 system behavior to entrainment is unaffected in lieu of the excessive amounts of liquid in the inner vessel during the tests referred to in the August 13, 2003 letter.

### **Westinghouse Response:**

The appropriate parameters for assessing the scaling of the downcomer liquid inventory are obtained from the governing conservation equations. The situation of particular interest is the liquid inventory depletion in the downcomer during the ADS-IRWST transition phase of a limiting SBLOCA such as a DEDVI event where downcomer liquid inventory is most seriously challenged.

### *Derivation of Scaling Parameters*

To obtain the appropriate scaling parameters, apply the conservation of mass equation to the downcomer region such that downcomer liquid inventory is depleted to satisfy core cooling and is not replenished via safety injection. The conservation of liquid mass in the downcomer region for this situation is as follows:

$$\frac{dM_{\text{downcomer liquid}}}{dt} = -\dot{m}_{\text{out}} = -\dot{m}_{\text{core}}$$

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

The liquid inventory can be represented via the liquid volume and density such that:

$$\rho_f \frac{dV_{\text{downcomer liquid}}}{dt} = -\dot{m}_{\text{core}}$$

Non-dimensionalizing the variables in the above equation and dividing by the mass-flow required to satisfy core cooling, the following downcomer liquid inventory scaling equation is obtained:

$$\left[ \frac{\rho_f \Delta V_{dc}}{m_{\text{core}}} \right]_{\text{ref}} \frac{dV_{\text{downcomer liquid}}^+}{dt} = -\dot{m}_{\text{core}}^+$$

The above equation can be re-expressed in terms of a time constant ( $\tau$ ) that represents the time to drain or deplete the downcomer liquid inventory to satisfy core cooling in the absence of safety injection to replenish the downcomer:

$$\tau \frac{dV_{\text{downcomer liquid}}^+}{dt} = -\dot{m}_{\text{core}}^+$$

where the time constant is defined as the liquid inventory storage relative to the depletion rate:

$$\tau = \left[ \frac{\rho_f \Delta V_{dc}}{m_{\text{core}}} \right]_{\text{ref}}$$

The appropriate scaling ratio for downcomer liquid inventory is therefore obtained by comparing the above time constant for the APEX-1000 test facility to AP1000:

$$\tau_{\text{Ratio}} = \frac{\left[ \frac{\rho_f \Delta V_{dc}}{m_{\text{core}}} \right]_{\text{APEX}}}{\left[ \frac{\rho_f \Delta V_{dc}}{m_{\text{core}}} \right]_{\text{AP1000}}}$$

The ideal time scaling ratio for APEX-1000 relative to AP1000 is ½. Ratios less than ½ indicate that APEX liquid inventory is depleted faster than AP1000 on a scaled basis, and vice-versa.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

### *Numerical Evaluation of Scaling Parameters*

The downcomer volume scaling ratio of APEX-1000 relative to AP1000 is about 1/112. The scaling of the downcomer in APEX results in a larger scaled volume relative to other reactor vessel volumes.

The core mass-flow ratio of APEX-1000 relative to AP1000 is about 1/58. This results in a larger scaled mass-flow rate in APEX-1000 relative to AP1000 and was obtained by applying the Simple Model to APEX-1000 and AP1000 at scaled power and downcomer level for a DEDVI event. The primary difference in inputs to the Simple Model was core inlet temperature (about 50 degrees additional subcooling for APEX) and backpressure where 14.7 psia is used for APEX-1000 (as only atmospheric backpressure has been tested at APEX) and 25 psia for AP1000. Applying these volume and massflow ratios, it can be seen that the downcomer drain time ratio between APEX-1000 and AP1000 is about 1/2, which is the ideal ratio.



Thus the APEX test facility is adequately scaled for downcomer inventory depletion relative to AP1000 during a potential situation in a SBLOCA where only the liquid inventory in the downcomer is available for core cooling.

### **Design Control Document (DCD) Revision:**

None

### **PRA Revision:**

None



# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number:** 21.5-2 Item 20

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

***Summary of Issue:***

In APEX-1000 Test DBA-02, Figure 21.5-2.26 (in Westinghouse letter DCP/NRC1611, August 13, 2001) shows the comparison of integrated vessel side break flow between the test data and NOTRUMP simulation.

Please explain why the integrated break flow test data decreases after about 1050 seconds.

**Westinghouse Response:**

Figure 1 shows the break flow for the three DEDVI tests, DBA-01, DBA-02 and DBA-03. For each test, the break flow is essentially the same through the blowdown, and falls to zero once the ADS valves provide a less restrictive flow path. Additional flow information after about 500 seconds is not considered relevant, and should not be used. The cause for these flow fluctuations may involve conditions in the break separator system, and there is no basis for these flow measurements to be considered valid.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



**Figure 1: Comparison of Break Flow For the DEDVI Tests**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

DSER Open Item Number: 21.5-2 Item 21

Original RAI Number(s): None

### *Summary of Issue:*

In the time period between approximately 350 and 1000 seconds for the DBA-02 test simulation, Figure 21.5-2.29 shows the core inlet temperature calculated by NOTRUMP to be about 25°F higher than data; whereas Figures 21.5-2.30 and 21.5-2.20 respectively show almost the same core outlet temperature and the core average void fractions between the test data and the NOTRUMP simulation.

Explain the apparent inconsistency between the test data and the NOTRUMP simulation in terms of energy balance.

### **Westinghouse Response:**

There is no inconsistency in energy balance because the higher NOTRUMP inlet temperature during this time period results in higher vapor generation in the core to remove the same amount of energy as is removed in the test. This effect can be seen in Figure 21.5-2.20 by the increase in NOTRUMP core void fraction between about 350 and 400 seconds, during which time the NOTRUMP simulation goes from under-predicting core inlet temperature to over-predicting core inlet temperature. The core exit temperature is nearly the same for the test and NOTRUMP during this time period because the core exit flow is at saturation conditions, and the NOTRUMP system pressure is about the same as in the test (Figure 21.5-2.5).

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



a,b,c

**Figure 21.5-2.20 Test DBA-02, Core Average Void Fraction**



a,b,c

**Figure 21.5-2.2 Test DBA-02, Downcomer Pressure Comparison**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number:** 21.5-2 Item 22

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

***Summary of Issue:***

For the NOTRUMP simulation of both DBA-02 and DBA-03 tests, Figures 21.5-2.19 and 21.5-2.44, respectively, show NOTRUMP overpredicts the core collapsed liquid level for the time period between about 100 to 400 seconds. Figures 21.5-2.20 and 21.5-2.45, respectively, indicate NOTRUMP underpredicts core average void fraction between the same time period. Westinghouse attributes the non-conservative NOTRUMP calculations to its lack of two-dimensional downcomer modeling and heating of DVI injection flow. The sensitivity study performed with higher injection temperature still show non-conservative, though reduced, NOTRUMP predictions (Figures 21.5-2.17 and 18).

Please explain why, with the same downcomer modeling deficiency, the NOTRUMP calculations of the core liquid level and void fraction are comparable with the tests after 400 seconds.

**Westinghouse Response:**

The NOTRUMP simulations are comparable with the DEDVI tests after 400 seconds because the test behavior is more consistent with NOTRUMP one dimensional modelling after this time. During the 100 to 350 second time period there is blowdown through the broken DVI nozzle on one side of the downcomer with high rates of injection of subcooled water from the intact DVI nozzle on the other side of the downcomer, This creates the two dimensional behavior observed in the tests during the 100 to 350 second time period. After ADS1 actuation, the break flow decreases as is shown in Figure 1. After 400 seconds the behavior is more one dimensional as the ADS4 flow paths become the dominant vent path and the injection rate is at the lower rate from the intact IRWST injection line.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

**AP1000 DESIGN CERTIFICATION REVIEW**  
**Draft Safety Evaluation Report Open Item Response**

---



**Figure 1: Comparison of Integrated Flow from the Break and ADS Flow Paths for DBA-02**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number:** 21.5-2 Item 23

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

***Summary of Issue:***

For DBA-03 test, Table 21.5-1.2 (DSER OI 21.5-2P Addendum 1 Page 18 in Westinghouse letter DCP/NRC1611, August 13, 2001) indicates that the intact accumulator injection starts at 110 seconds and empties at 510 seconds for the test data, and starts at 123 seconds and empties at 346.44 seconds for the NOTRUMP simulation. Page 9 states that the comparison for accumulator 2 [Intact] is considered minimal, and that the minimal prediction is considered to have a negligible impact on the results as the composite effect of CMT and accumulator injection is reasonably/conservatively predicted by NOTRUMP.

Please explain the cause of the large difference between the NOTRUMP simulation and test data with regard to the accumulator injection starting and empty times.

**Westinghouse Response:**

The NOTRUMP simulation predicts a discrete accumulator injection period to occur which slows CMT draining. Once the accumulator empties, CMT injection becomes the sole makeup source until IRWST injection. This is consistent with the observed response for APEX-1000 Test DBA-02. Test DBA-03 indicates a larger interaction between the CMT and accumulator than predicted by NOTRUMP or observed in Test DBA-02. This extended accumulator injection period in Test DBA-03 results in less vessel inventory depletion by providing more continuous injection. The NOTRUMP simulation behavior results in the prediction of an injection gap and subsequently reduced vessel inventory compared to the test simulation which is considered to be conservative.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None



# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

DSER Open Item Number: 21.5-2 Item 24

Original RAI Number(s): None

### *Summary of Issue:*

It is noted that several different NOTRUMP model approaches are used in the NOTRUMP simulation of the APEX-1000 tests compared to the AP1000 small break LOCA DBA analysis. These include revised noding in the pressurizer, revised noding in the core makeup tank, no PRHR heat transfer area adjustment is applied, use of different ADS critical and non-critical flow models, and use of a break flow multiplier to more accurately represent the results observed in the test.

Please provide a complete list of all differences of the NOTRUMP model between the APEX-1000 simulation and AP1000 plant analyses, and justify why the test simulation conclusions can be applied to the NOTRUMP models used for the AP1000 analysis.

### **Westinghouse Response:**

As described in Section 1.16 of Reference 1, the basic reactor coolant system (RCS) noding utilized is an evolution of the standard plant NOTRUMP Evaluation Model (EM). Specific noding was added for the passive safety system components, such as the passive residual heat removal (PRHR) heat exchanger, core makeup tanks (CMTs), in-containment refueling water storage tank (IRWST), and automatic depressurization systems (ADS). The noding bases for these individual components were generated by analysis of individual separate effect tests (SET) for each component. The noding for the individual components obtained from the SETs was then applied to the integral effects test (IET) i.e., SPES-2 and Oregon State University (OSU), and eventually to the AP600/AP1000 plant models themselves.

The basic rule applied was to preserve the number of fluid volumes utilized in each main component utilized in the IET and SET facility simulations. The exception to this rule was described in Reference 1 along with the results of sensitivity studies performed on various components and/or locations. The specific sensitivities are mentioned below:

- Core Noding Sensitivity Studies
- PRHR Noding Sensitivity Studies
- CMT Noding Sensitivity Studies
- Downcomer Noding Sensitivity Studies

The changes in noding/methodology between the AP600 and AP1000 APEX simulations were described in Reference 2, Section E1.0. In addition, the additional conservatisms applied to the Advanced Plant (either AP600 or AP1000) methodology are described in Section 1.17 of Reference 1. Table-1 contains a list of the modeling differences between the AP1000 APEX test facility and the AP1000 plant model.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

The most significant difference between the AP1000 plant and OSU simulations is the breakflow transition methodology utilized as described in Section E1.0 of Reference 2. This difference in methodology was driven by the hardware differences between the original AP600 APEX facility and the AP1000 APEX facility. The revised methodology employed utilizes the minimum flow predicted by the critical flow model and the orifice equation during and after the transition to non-critical flow as opposed to the application of the FLOAD4 resistance increase methodology. The FLOAD4 model, in its current form, is not appropriate for modeling the AP1000 APEX ADS-4 flow paths. As such, the revised methodology was implemented.

As described in Section E-1.0, sensitivity studies were performed with the revised methodology on the AP600 APEX facility for the DEDVI line simulation (Test SB12) and shown to provide similar response to that obtained with the momentum flux adjustment factor. Additionally, sensitivity studies were also performed for the AP1000 DEDVI line break. As was observed for the AP600 APEX simulations, the alternate methodology was shown to provide comparable results. A review of Figure 21.5-2.I24-1 through Figure 21.5-2.I24-3 demonstrates that the revised methodology (labeled IUNCHFL 4 in the figures) is less limiting for the AP1000 plant design resulting in increased RCS depressurization and ADS-4 discharge while negligibly impacting core coverage. As such, the use of the FLOAD4 methodology on the plant design is considered to be conservative.

The PRHR heat transfer area was not adjusted for the AP1000 APEX simulation as was the case for the AP1000 plant model. The purpose of the heat transfer modifications to the AP1000 plant model is to assure conservative behavior of the heat removal system. A conservative prediction of PRHR heat transfer (i.e. under-prediction) requires the other passive components (i.e. CMT, ACC, ADS) and break to offset the additional energy not being removed by the PRHR. This is considered to be a conservative feature in the AP1000 plant simulations.

### References

1. WCAP-14807, Revision 5, "NOTRUMP Final Validation Report for AP600," August 1998.
2. WCAP-15644-P, Revision 1, "AP1000 Code Applicability Report," September 2003.

### Design Control Document (DCD) Revision:

None

### PRA Revision:

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

**Table-1 Model Feature Comparison**

Feature	APEX-1000 Model	AP1000 Plant	Discussion
<b><i>Noding Related</i></b>			
CMT Noding	19-node CMT	4-node CMT	See Section 1.17 of Reference 1 and Section E1.0 of Reference 2
Pressurizer Noding	4-node pressurizer	1-node pressurizer	See Section E1.0 of Reference 2
Downcomer Noding	1-node downcomer	3-node downcomer	See Section 1.17 of Reference 1
PRHR Noding	7-nodes	9-nodes	See Section 1.17 of Reference 1
Core Noding	4-nodes	14-nodes	See Section 1.17 of Reference 1
<b><i>Passive System Flow Resistances</i></b>	Nominal	Maximum	See Section 1.18 of Reference 1
<b><i>PRHR Heat Transfer Area</i></b>	Nominal	Reduced by 50%	Required to assure conservative PRHR heat transfer. See above.
<b><i>Core Power</i></b>	Test values as scaled to AP1000.	ANS 71+20%	Appendix-K Required
<b><i>Break Modeling</i></b>			
Break Model	Fauske/HEM	Moody	Appendix-K Required. See RAI-440.721-J of Reference 1
ADS Discharge Paths Break Model	Fauske/HEM	Fauske/HEM	Appendix-K Required. See RAI-440.721-J of Reference 1
ADS Breakflow Transition Model	Minimum of Critical and Orifice Equation	FLOAD4 Resistance Adjustment	Utilized to compensate for NOTRUMP lack of detailed momentum flux model
Break Discharge Coefficient	0.7	Nominal	Utilized to achieve proper match to test data.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

### AP1000 DEDVI Re-noded IUNCHFL=4 Sensitivity Study Core/Upper Plenum Mixture Level

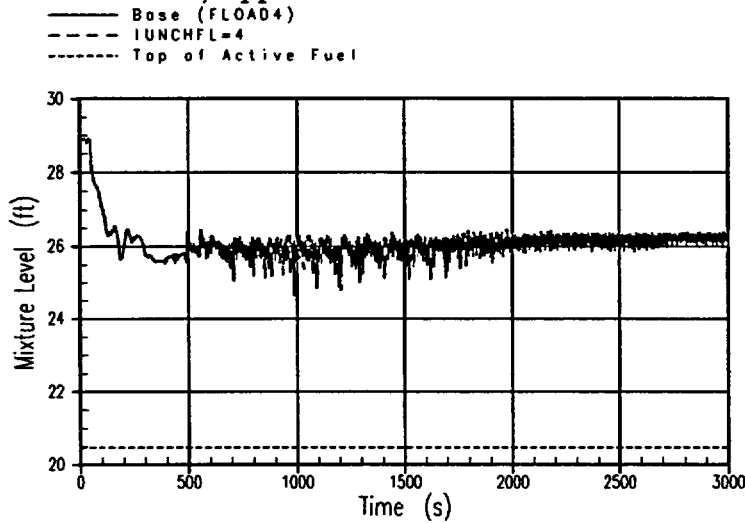


Figure 21.5-2.I24-1 AP1000: Core Upper Plenum Two-Phase Mixture Level

### AP1000 DEDVI Re-noded IUNCHFL=4 Sensitivity Study Downcomer Pressure at DVI Port

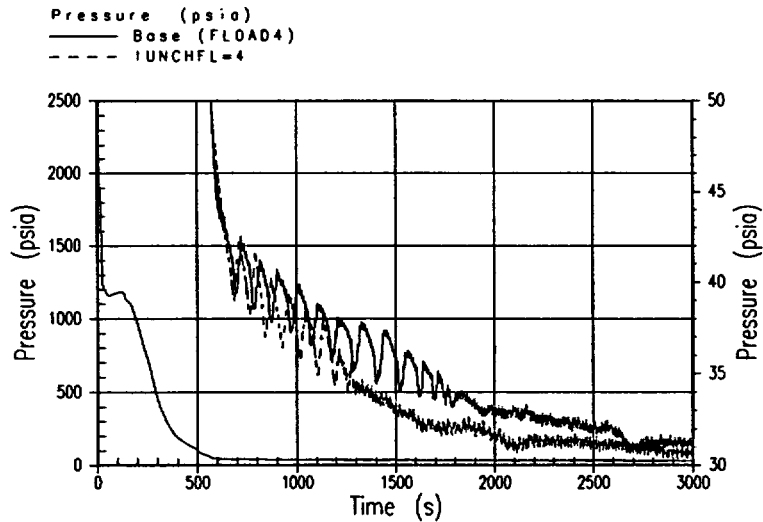


Figure 21.5-2.I24-2 AP1000: Downcomer Pressure At DVI Injection Port

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

### AP1000 DEDVI Re-noded IUNCHFL=4 Sensitivity Study Integrated ADS-4 Discharge

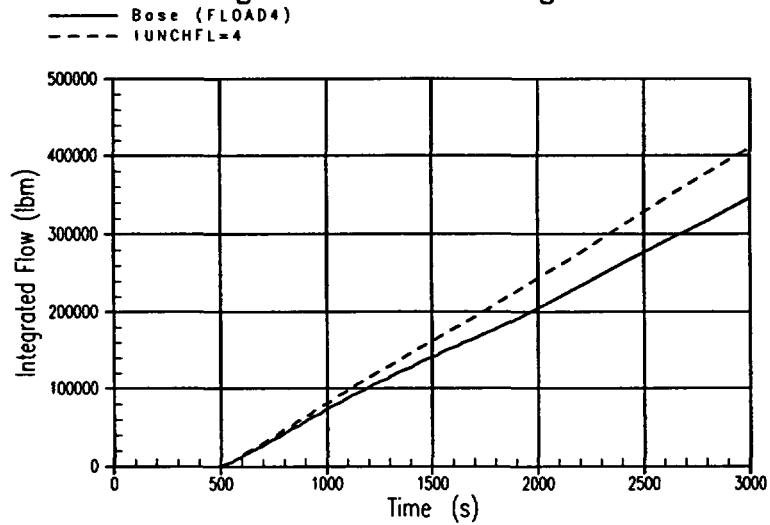


Figure 21.5-2.I24-3 AP1000: Integrated ADS-4 Discharge

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number:** 21.5-2 Item 25

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

***Summary of Issue:***

For the test DBA-04, 2-inch cold leg break simulation (Test Acceptance Report OSU-AP1000-04), Table 4-1 indicates that the assumed single failure is failure of 1 of 2 lines in one ADS-4 train on the pressurizer side, whereas Section 5.0, "Test Procedure," states that the 100-percent flow nozzle was installed in the ADS 4-2 (on hot leg 2) and the 50-percent flow nozzle was installed in ADS 4-1 (on hot leg 1). Since the pressurizer is on hot leg 2, the use of 50-percent flow nozzle on hot leg 1 appears to simulate a single failure of ADS-4 on the non-pressurizer side, contradictory to Table 4-1.

- A. Please clarify this inconsistency
- B. The results of the DBA-02 and DBA-03 tests simulating DVI line break indicate that the single failure of ADS-4 valve on the non-pressurizer side is limiting. Please explain why a single failure of ADS-4 valve on the pressurizer side was simulated for the DBA-4 2-inch cold leg break test.

**Westinghouse Response:**

There is an error in the report. For DBA-04, Table 4-1 should indicate that the assumed single failure of 1 of 2 lines in one ADS-4 train is on the non-pressurizer side. The information is correct in Section 5.0. Westinghouse will issue errata for this report to correct this error.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

**AP1000 DESIGN CERTIFICATION REVIEW**  
**Draft Safety Evaluation Report Open Item Response**

---

**DSER Open Item Number: 21.5-2 Item 26**

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

***Summary of Issue:***

For the test TR-02-D, a 2-inch cold leg break simulation with ADS-4 actuation at plant-prototypic pressure conditions (Test Acceptance Report OSU-AP1000-05), Table 4-1 indicates the assumed failure of one ADS-4 train and no ADS 1-3. However, Section 5, A Test Procedure, of the report indicates that flow nozzles that simulate full flow for ADS-1, 2, and 3 were installed.

Please clarify whether the ADS-1, 2, and 3 valves were open during the test, and whether this is consistent with Table 4-1, which indicates no ADS-1, 2, and 3 for the test.

**Westinghouse Response:**

There is an error in the report. ADS-1, 2, and 3 valves were open during test TR-02-D. Westinghouse will issue errata for this report to address this error.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number: 21.5-2 Item 27**

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

***Summary of Issue:***

In its August 13, 2003, letter (DCP/NRC-1611), Westinghouse provided in Response Addendum 1 the NOTRUMP simulation of APEX-AP1000 tests DBA-02 and DBA-03.

Please provide the NOTRUMP simulation of other APEX-AP1000 tests.

**Westinghouse Response:**

WCAP-15644-P Revision 1 was submitted by Westinghouse letter DCP/NRC1627 dated September 19, 2003. This revision incorporates (in Appendix E) the NOTRUMP simulation of the APEX-1000 tests referred to in this NRC comment. Section 3 of WCAP-15644-P Revision 1 summarizes the NOTRUMP validation for small break LOCA analysis, much of which is derived from comparison to AP600 tests that have been shown to scale adequately to AP1000 (WCAP 15613, "AP1000 PIRT and Scaling Assessment," February 2001). Section 3.5 of WCAP-15644-P Revision 1 summarizes how the additional AP1000 NOTRUMP verification/simulation items raised in the AP1000 Design Certification review have been addressed. An additional NOTRUMP simulation of APEX-1000 test NRC-AP1000-05 is provided in the response to NRC comment 30.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None



# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number: 21.5-2 Item 28**

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

### ***Summary of Issue:***

The APEX test matrix contained a subset of tests expected to produce the lowest vessel collapsed liquid levels. The selection of these tests was in part based on results of NOTRUMP simulations. Westinghouse indicated that similar liquid levels were predicted for inadvertent ADS 1-3, 2 inch hot leg break and the DEDVI break. The DEDVI break is considered to be limiting small break LOCA and assessment has focused on this case. Please demonstrate the adequacy of the NOTRUMP simulation of inadvertent ADS 1-3 to ensure that the limiting break has been identified.

### **Westinghouse Response:**

The APEX-600 test matrix included a spectrum of break sizes and locations. The APEX-600 tests have been shown to scale adequately to AP1000 (WCAP 15613, "AP1000 PIRT and Scaling Assessment," February 2001). The NOTRUMP simulations of the APEX-600 tests, including the inadvertent ADS and DEDVI tests, are reported in WCAP-14807-P Revision 5, August 1998, "NOTRUMP Final Validation Report for AP600". Comparison of the NOTRUMP simulation core collapsed liquid level to test data for these two tests is reproduced in Figures 28-1 and 28-2. The APEX-600 DEDVI test exhibited a lower core collapsed liquid level than did the APEX-600 Inadvertent ADS test, although both tests exhibited adequate core cooling throughout. The NOTRUMP simulations for these two tests exhibit this trend of lower core collapsed liquid level for the DEDVI case as compared to the Inadvertent ADS case. The NOTRUMP simulations for APEX-1000 DEDVI tests are reported in WCAP-15644-P Revision 1, submitted by Westinghouse letter DCP/NRC1627 dated September 19, 2003, and show similar comparison to test data as for the APEX-600 DEDVI case. This provides confidence that the NOTRUMP analysis of the AP1000 small break LOCA spectrum, including the Inadvertent ADS case, are adequate.

### **Design Control Document (DCD) Revision:**

None

### **PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

a,b,c

[Empty response area]

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

[Empty response box]

a,b,c

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

DSER Open Item Number: 21.5-2 Item 29

Original RAI Number(s): None

### Summary of Issue:

The Cunningham-Yeh correlation as described on OI 21.5-3 Page 3 has an error in the critical bubble radius term  $R_{bcr}$ .

Please confirm that it is only a typographic error and the correct Cunningham-Yeh correlation is used in the study.

### Westinghouse Response:

The Cunningham-Yeh correlation as described on OI 2.5-3 Page 3 did contain a typographical error as the item 29 asserts. The corrected equation is shown below:



a,c

### Design Control Document (DCD) Revision:

None

### PRA Revision:

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---

**DSER Open Item Number: 21.5-3 Item 30**

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

### ***Summary of Issue:***

Westinghouse has relied on comparison to data to validate NOTRUMP's modeling of level swell and upper plenum entrainment phenomena during the ADS-4/IRWST injection transition phase. These phenomena are high ranked during this phase because heat transfer from the rods to the coolant depends strongly on the proximity of the two-phase level and the amount of droplets in the dispersed flow heat transfer regime. As a result, the staff is thoroughly reviewing this information. If the NOTRUMP code were shown to conservatively predict rod heat up for appropriately scaled test cases run at APEX-1000 that demonstrated heat up, such as NRC-AP1000-05, then the uncertainty in the staff's review of the effect of level swell could be reduced.

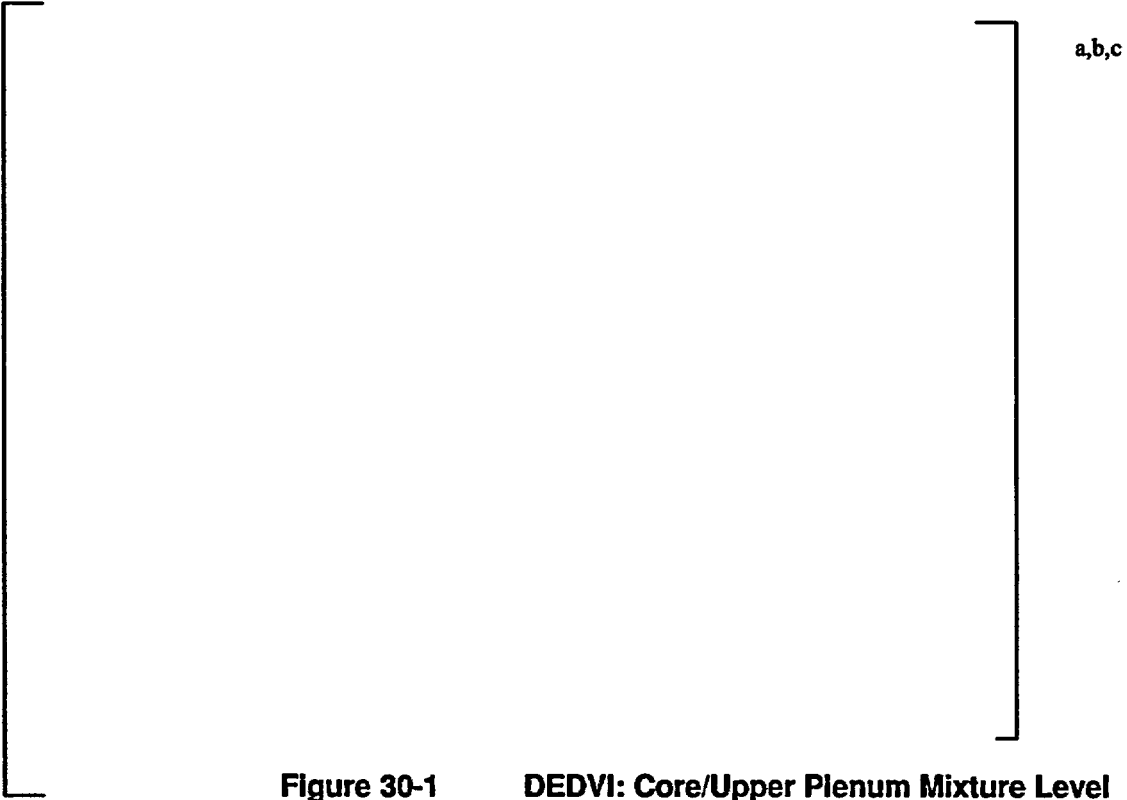
### **Westinghouse Response:**

WCAP-15644-P Revision 1 was submitted by Westinghouse letter DCP/NRC1627 dated September 19, 2003. This revision incorporates (in Appendix E) the NOTRUMP simulation of APEX-1000 tests DBA-02 and DBA-03. An additional NOTRUMP simulation of APEX-1000 has been performed that we believe is representative of NRC-AP1000-05. This simulation was performed using the same NOTRUMP input as for the DBA02 simulation except that the non-pressurizer side ADS4 path was assumed not to open during the test. This represents a beyond design basis situation for AP1000 and represents a failure scenario in the AP1000 PRA. The results of this NOTRUMP simulation are shown in Figures 30-1 through 30-10. The main result is that NOTRUMP indicates core uncover and heatup at about the same time as in the test, based on the NRC-AP1000-05 test results presented by NRC in the July 16-17, 2003, ACRS T/H subcommittee meeting.

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



**Figure 30-1 DEDVI: Core/Upper Plenum Mixture Level**



**Figure 30-2 DEDVI: Upper Plenum Vapor Temperature**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



a,b,c

**Figure 30-3 DEDVI: Downcomer Two-Phase Mixture Level**



a,b,c

**Figure 30-4 DEDVI: Downcomer Pressure at DVI Port**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



a,b,c

**Figure 30-5 DEDVI: Intact DVI Line Injection Flow**



a,b,c

**Figure 30-6 DEDVI: Core Collapsed Liquid Level**



# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



a,b,c

**Figure 30-7 DEDVI: Pressurizer Two-Phase Mixture Level**



a,b,c

**Figure 30-8 DEDVI: Pressurizer Pressure**

# AP1000 DESIGN CERTIFICATION REVIEW

## Draft Safety Evaluation Report Open Item Response

---



a,b,c

**Figure 30-9      DEDVI: ADS 4 Integrated Liquid Discharge**



a,b,c

**Figure 30-10      DEDVI: ADS 4 Integrated Vapor Discharge**

**AP1000 DESIGN CERTIFICATION REVIEW**  
**Draft Safety Evaluation Report Open Item Response**

---

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None