Westinghouse

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

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

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

Your ref: Docket No. 52-006
Our ref: DCP/NRC1611

August 13, 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. Please note that Enclosure 1 to this letter supercedes Enclosure 1 to Westinghouse letter DCP/NRC1610 dated August 7, 2003.

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.

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August 13, 2003

The Westinghouse Electric Company Application for Withholding and Affidavit are also attached to this submittal letter as Enclosure 1. Attachment 2 contains Westinghouse proprietary information consisting of trade secrets, commercial information or financial information which we consider privileged or confidential pursuant to 10 CFR 2.790. Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosures. Attachment 3 contains no proprietary information.

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

Correspondence with respect to the application for withholding should reference AW-03-1687, and should be addressed to Hank A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania, 15230-0355.

Please contact me at 412-374-5355 if you have any questions concerning this submittal.

Very truly yours,

M. M. Corletti

M. M. Corletti

Passive Plant Projects & Development
AP600 & AP1000 Projects

/Enclosure

1. Westinghouse Electric Company Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit AW-03-1687.

/Attachments

1. List of the AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses transmitted with letter DCP/NRC1611
2. Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated August 13, 2003
3. Non-Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated August 13, 2003

August 13, 2003

			Attachment 2	Attachment 3
bcc:	Document Control	- US NRC, Rockville, MD	1P	1NP
	C. B. Brinkman*	- Westinghouse, Rockville, MD		
	M. M. Corletti	- Westinghouse, Pittsburgh, PA, EC E3	2P	2NP
	W. E. Cummins*	- Westinghouse, Pittsburgh, PA, EC E3		
	J. Segala	- US NRC, Rockville, MD	6P	6NP
	H. A. Sepp*	- Westinghouse, Pittsburgh, PA, EC E4-07A		
	R. P. Vijuk*	- Westinghouse, Pittsburgh, PA, EC E3-05		
	J. W. Winters*	- Westinghouse, Pittsburgh, PA, EC E3-08		

* without attachments

DCP/NRC1611
Docket No. 52-006

August 13, 2003

Enclosure 1

Westinghouse Electric Company
Application for Withholding and Affidavit

Westinghouse

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

August 13, 2003

AW-03-1687

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-03-1687 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-03-1687 and should be addressed to the undersigned.

Very truly yours,

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

/Enclosures

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AW-03-1687

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared James W. Winters, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company, LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief.

James W. Winters

James W. Winters, Manager
Passive Plant Projects & Development
Nuclear Power Plants Business Unit
Westinghouse Electric Company, LLC

Sworn to and subscribed
before me this 13th day
of August, 2003

Lorraine M. Piplica

Notary Public

LORRAINE M. PIPLICA
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

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

(1) I am Manager, Passive Plant Projects & Development, in the Nuclear Power Plants Business Unit, of the Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company, LLC.

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

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company, LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

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

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

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

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

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

This information is being transmitted by Westinghouse's letter and Application for Withholding Proprietary Information from Public Disclosure, being transmitted by Westinghouse Electric Company (W letter AW-03-1687) and to the Document Control Desk, Attention: John Segala, DIPM/NRLPO, MS O-4D9A.

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

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

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

Further this information has substantial commercial value as follows:

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

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

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

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

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

Further the deponent sayeth not.

DCP/NRC1611
Docket No. 52-006

August 13, 2003

Attachment 1

List of

Proprietary and Non-Proprietary Responses

Table 1	
"List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1611"	
3.6.3.4-2P* 3.6.3.4-2 6.2.1.8.3-1, Rev 1 6.2.1.8.2-1, Rev. 1 6.2.1.8.3-3, Rev. 1 9.5.1-2, Rev. 1 21.5-2P* 21.5-2 21.5-2P Addendum 1* 21.5-2 Addendum 1 * Westinghouse Proprietary	

DCP/NRC1611
Docket No. 52-006

August 13, 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: **3.6.3.4-2**

Original RAI Number(s): **251.004**

Summary of Issue:

In RAI 251.005, the staff requested that the applicant provide values of crack morphology parameters, e.g., surface roughness, number of 45 degree and 90 degree turns, etc., that were used in generating the BACs for LBB. The NRC staff also asked for a comparative study, using the values of crack morphology parameters associated with transgranular stress corrosion cracking (TGSCC). This information and the study were requested to evaluate the BACs and to understand the sensitivity of the AP1000 LBB analyses to a crack morphology similar to PWSCC. In its response to RAI 251.005, the applicant provided the values of crack morphology parameters used in generating the BACs. However, since chlorides will be controlled at minimum levels in the AP1000 LBB candidate piping systems water environment and the hydrogen overpressure will keep the oxygen levels to near zero, the applicant discounted the possibility of TGSCC and considered the comparative study using the crack morphology parameters associated with TGSCC not necessary. The applicant's argument does not address the intent of RAI 251.005. The NRC staff performed an independent sensitivity study to assess the impact on the BACs due to a consideration of a TGSCC type of crack in the LBB analysis as a surrogate for PWSCC. The NRC staff's independent sensitivity study shows that the BACs might not be easily met by the most limiting piping. DCD Tier 2 Appendix 3B.3.3.4 does not rule out the possibility of a LBB candidate piping system not meeting the BAC limit either, as evidenced by the statement: "[i]f the point falls above the bounding analysis curve, the leak-before-break analysis criteria are not satisfied and the pipe layout or support configuration needs to be revised to meet the leak-before-break bounding analysis."

The information provided by the applicant has not been sufficient to address the staff position in SECY-93-087, discussed in DSER Section 3.6.3.1, on demonstrating that adequate margins on leakage, loads, and flaw sizes are available for AP1000 LBB candidate piping systems. In addition, the information provided is not sufficient to understand the degree to which PWSCC may affect LBB margins. Therefore, the staff is evaluating the appropriate analyses the applicant should perform to resolve these issues. The staff expects to issue a supplemental DSER on LBB. This is Open Item 3.6.3.4-2.

Westinghouse Response:

In order to avoid the concern of the Inconel 82/182 PWSCC issue, Inconel 52/152 will be used in the AP1000 for all the applicable locations in LBB piping systems. Transgranular stress corrosion cracking (TGSCC) has not been observed in PWR stainless steel piping. For TGSCC to occur, an aggressive chemical species, such as chlorides, would also need to be present. Since these will be controlled and kept at minimum levels in the water environment of AP1000 LBB candidate piping systems, a much higher level of oxygen would be required than is present

in the primary coolant to provide the appropriate environment for SCC to develop. Since the oxygen levels are kept to near zero by the hydrogen overpressure, the occurrence of TGSCC is highly unlikely.

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

Westinghouse believes that Transgranular Stress Corrosion Cracking (TGSCC) is not a credible mechanism for the AP1000 LBB candidate piping systems. However, as requested by the NRC, a sensitivity study using hypothetical crack morphology parameters due to IGSCC/TGSCC has been conducted, and the results are reported herein.

The NRC RAI 251.005 has indicated that the information regarding the crack morphology parameters for various degradation mechanisms is available in NUREG/CR-6443 (Reference 1).

Westinghouse has reviewed the information presented in NUREG/CR-6443. In section 3.3 of NUREG/CR-6443, a leak rate sensitivity study is described using various crack morphologies. Table 3-1 of NUREG/CR-6443 for evaluations by both SQUIRT (developed by Battelle) and PICEP (Reference 2, developed by EPRI) presents summaries of surface roughness for IGSCC and corrosion-fatigue by Battelle. There are no references provided for the roughness and number of turns used for either case. The parameters used by Battelle in the sensitivity study using PICEP do not agree with the recommendations from the PICEP manual except for the surface roughness for IGSCC. The number of 90-degree (48) turns for IGSCC for PICEP used in the study is 4 times the PICEP manual recommendation.

Westinghouse has reviewed the information presented in NUREG/CR-6300 (Reference 3.15 of NUREG/CR-6443) to determine the source of the roughness and number of turns in the series of reports produced by Battelle. The crack morphology parameters presented in NUREG/CR-6300 are based on global and local surface roughness and the local number of turns. In NUREG/CR-6300, equations 9-1 and 9-2 are presented by Battelle to calculate the equivalent surface roughness and number of turns based on the COD (Crack Opening Displacement). Battelle has also indicated that " the piecewise linear variation of the crack morphology variables is a first attempt to simulate their dependency on COD. The numerical constants in equations 9-1 and 9-3 are based on a review of cracks found in service and expert opinion by Battelle,". The Battelle report concludes that "additional studies are needed to evaluate these ... models,". However, Westinghouse has used these two equations for the sensitivity study.

It is important to know that the data presented in NUREG/CR-6300 are all for IGSCC in BWR piping. These data are not applicable for the AP1000 piping systems. The number of turns and global and local surface roughness presented is highly subjective. The NRC has requested a sensitivity study for the AP1000 DVI-A system using the TGSCC crack morphology as a surrogate for PWSCC. From the examination of the actual crack shape (Reference 3, Figure 5

for TGSCC branching in Ft. Calhoun Type 348 stainless steel failed CEDM housing) it can be concluded that the crack morphology parameters for TGSCC are less severe than the IGSCC parameters presented in NUREG/CR-6300. Therefore, the leakage flow will be higher for TGSCC than for IGSCC, i.e., the leakage flaw size for a given leak rate will be smaller for TGSCC than for IGSCC. There are no crack morphology parameters (surface roughness and number of turns) readily available for IGSCC mechanism. However the highly conservative, hypothetical IGSCC crack morphology parameters shown in NUREG/CR-6300 are used for the AP1000 sensitivity study.

To evaluate the effects of surface roughness and number of turns in the AP1000 sensitivity study, the data from Tables 9.1 and 9.3 of NUREG/CR-6300 were used for the 6-inch diameter and a limiting 8-inch diameter DVI-A piping system. In addition, the recommendations from the PICEP manual for IGSCC cracking were evaluated, i.e. a surface roughness of 200 micro-inches combined with twenty-four 45-degree turns per inch of thickness. The PICEP leak rate prediction recommendations for IGSCC were validated against actual leakage data. There is no evidence that the surface roughness and number of turns presented in NUREG/CR-6300 have been validated with actual test data. The only recommendation in NUREG/CR-6300 is that "....further studies are needed to verify these results."

The number of 90-degree turns per inch for NP-2472, Vol. 2 (Figure 1-1) and NP-3684SR, Vol. 3 (paper 4, Figure 5) cases are 1450 and 873 as shown in Table 9.3 of NUREG/CR-6300. Westinghouse believes these numbers are extremely conservative and should not be used in this study. However, for reference purpose only, calculations were also performed using these two extremely conservative cases.

As mentioned above, IGSSC/TGSCC in the AP1000 piping is not expected to occur. The sensitivity study with hypothetical IGSCC cracking morphology are performed for a reference flaw size (1.5 x leakage flaw size) which is the average of the leakage flaw size and the critical flaw size. The leakage flaw size referred to in this sensitivity study is half of the critical flaw size. The sensitivity study was performed for the DVI-A system with actual AP1000 pipe stress analysis loads for the 6-inch (Line Number L019A) line and a limiting 8-inch (Line Number L027A) line.

Table 1 shows the IGSCC crack morphology parameters obtained from NUREG/CR-6300. Table 2 and Table 3 show the effective roughness and the effective number of turns for 8-inch line. Table 4 shows the predicted leakage and margin on leak rate for a reference flaw size of 1.5 x leakage flaw size for the 8-inch line. Table 5 and Table 6 show the effective roughness and the effective number of turns for 6-inch line. Table 7 shows the predicted leakage and margin on leak rate for a reference flaw size of 1.5 x leakage flaw size for the 6-inch line.

Excluding the cases of NP-2472, Vol. 2 (Figure 1-1) and NP-3684SR, Vol. 3 (paper 4, Figure 5) cases which are extremely conservative, the margins on leak rate for a 8-inch line with hypothetical IGSCC varies from 6.1 to 35 and the margins on leak rate for a 6-inch line with hypothetical IGSCC varies from 13.6 to 74.4. Also shown in these tables are the results for the normal case (fatigue crack). For the AP1000 the leak detection capability is 0.50 gpm. For the

AP1000, LBB analyses Westinghouse has used a surface roughness of [ ]^{a,c,e} micro-inches, which is higher (more conservative) than the industry-recommended value.

In conclusion, the sensitivity study demonstrates that there are ample margins on leak detection for the AP1000 piping systems with an extremely low probability IGSCC cracking. The leakage from these extremely low probability crack morphologies can therefore be detected.

REFERENCE

1. NUREG/CR-6443, "Deterministic and Probabilistic Evaluations for Uncertainty in Pipe Fracture Parameters in Leak-Before-Break and In-Service Flaw Evaluations."
2. EPRI-NP- 3596-SR, "PICEP: Pipe Crack Evaluation Computer Program," Revision 1, July 1987.
3. Sixth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactor, Paper Entitled: "Evaluation of Cracking In Type 348 Stainless Steel Control Element Drive Mechanism Housings", B. Lisowyj, August 1-5, 1993, San Diego, CA pp. 343-350.

Table 1.
IGSCC Crack Morphology

Source	Roughness (micro-inches)		90-degree Turns (# per inch)
	Local	Global	
IGSCC Values from NUREG/CR-6300, Table 9.1 and Table 9.3			
NP-2472 Vol. 2 (see Figure I-1)	80	3,990	1,450
NP-3684SR, Vol. 3 (Paper 4, Figure 11)	290	2,930	352
NP-3684SR, Vol. 3 (Paper 4, Figure 5)	412	1,650	873
NP-3684SR, Vol. 2 (Paper 5, Figure 21)	25.0 to 250	5,000	240
NP-3684SR, Vol. 2 (Paper 19, Figure12)	55	1,100	670

Note: IGSCC parameters recommended by PICEP are surface roughness of 200 micro-inches and twenty-four 45 degree turns per inch (which is equivalent to twelve-90 degree turns per inch).

Table 2.
Calculated Effective Roughness for 8-inch Line

Case	Roughness, micro-inches 1.5 x Leakage Flaw Size (8.64")
Normal Case (Fatigue Crack)	[ ][a,c,e]
PICEP Recommended IGSCC Crack	200
Cases from NUREG/CR-6300	
NP-2472 Vol. 2 (Figure I-1)	996
NP-3684SR Vol. 3 (Paper 4, Figure 11)	1142
NP-3684SR Vol. 3 (Paper 4, Figure 5)	1131
NP-3684SR Vol. 2 (Paper 5, Figure 21)	1087
NP-3684SR Vol. 2 (Paper 19, Figure 12)	970

Table 3.
Effective Number of Turns, 8-inch Line

Case	Number of 90-degree Turns 1.5 x Leakage Flaw Size (8.64")
Normal Case (Fatigue Crack)	0
PICEP Recommended IGSCC Morphology	10
Cases from NUREG/CR-6300	
NP-2472 Vol. 2 (Figure I-1)	935
NP-3684SR Vol. 3 (Paper 4, Figure 11)	204
NP-3684SR Vol. 3 (Paper 4, Figure 5)	341
NP-3684SR Vol. 2 (Paper 5, Figure 21)	163
NP-3684SR Vol. 3 (Paper 19, Figure 12)	116

Table 4.
Predicted Leakage and Margins on Leak Rate for 8-inch Line

Case	Leakage for 1.5 x Leakage Flaw Size (gpm)	Margin on Leak Rate
Normal Case (Fatigue Crack)	36.75	73.5
PICEP Recommended IGSCC Morphology	17.52	35
Cases from NUREG/CR-6300		
NP-2472 Vol. 2 (Figure I-1)	1.49	3
NP-3684SR Vol. 3 (Paper 4, Figure 11)	3.07	6.1
NP-3684SR Vol. 3 (Paper 4, Figure 5)	2.38	4.8
NP-3684SR Vol. 2 (Paper 5, Figure 21)	3.47	6.9
NP-3684SR Vol. 2 (Paper 19, Figure 12)	4.22	8.4

Table 5.
Calculated Effective Roughness for 6-inch Line

Case	Roughness, micro-inches 1.5 x Leakage Flaw Size (6.6")
Normal Case (Fatigue Crack)	[ ]a,c,e
PICEP Recommended IGSCC Crack	200
Cases from NUREG/CR-6300	
NP-2472 Vol. 2 (Figure I-1)	1306
NP-3684SR Vol. 3 (Paper 4, Figure 11)	1427
NP-3684SR Vol. 3 (Paper 4, Figure 5)	1369
NP-3684SR Vol. 2 (Paper 5, Figure 21)	1391
NP-3684SR Vol. 2 (Paper 19, Figure 12)	1100

Table 6.
Effective Number of Turns, 6-inch Line

Case	Number of 90-degree Turns
	1.5 x Leakage Flaw Size (6.6")
Normal Case (Fatigue Crack)	0
PICEP Recommended IGSCC Morphology	8
Cases from NUREG/CR-6300	
NP-2472 Vol. 2 (Figure I-1)	678
NP-3684SR Vol. 3 (Paper 4, Figure 11)	140
NP-3684SR Vol. 3 (Paper 4, Figure 5)	173
NP-3684SR Vol. 2 (Paper 5, Figure 21)	121
NP-3684SR Vol. 3 (Paper 19, Figure 12)	44

Table 7.
Predicted Leakage and Margins on Leak Rate for 6-inch Line

Case	Leakage for 1.5 x Leakage Flaw Size (gpm)	Margin on Leak Rate
Normal Case (Fatigue Crack)	73.77	147.5
PICEP Recommended IGSCC Morphology	37.19	74.4
Cases from NUREG/CR-6300		
NP-2472 Vol. 2 (Figure I-1)	3.24	6.5
NP-3684SR Vol. 3 (Paper 4, Figure 11)	6.94	13.9
NP-3684SR Vol. 3 (Paper 4, Figure 5)	6.32	12.6
NP-3684SR Vol. 2 (Paper 5, Figure 21)	7.51	15
NP-3684SR Vol. 2 (Paper 19, Figure 12)	12.99	26

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 6.2.1.8.2-1 (Response Revision 1)

Original RAI Number(s): 650.004

Summary of Issue:

The applicant's February 21, 2003, response to RAI 650.004 also included an analysis of the IRWST screens' capability to accommodate debris accumulation. The staff's review of the applicant's analysis showed that the mass of resident debris assumed by the applicant (i.e., 227 kg, or 500 lb) was consistent with estimates made for current generation PWRs in the Generic Safety Issue (GSI) 191 parametric study (NUREG/CR-6772). However, the staff could not accept this analysis, primarily because the applicant assumed that a single density value is valid for all density-dependent calculations involving resident fibrous debris. According to the physical properties of analyzed types of fibrous materials, potentially different density values may be required to correctly determine the settling velocity (i.e., the material density), to calculate a volume from the assumed mass (i.e., the "as-found" density), and to determine the thickness and porosity of the associated debris bed (i.e., the rubblized density). As a result of the applicant's single-density assumption, which deviated significantly from the material properties of the low-density fiberglass on which the head loss data referenced by the applicant was based, the NRC staff concluded that the calculation was unacceptable. During a teleconference on April 3, 2003, the applicant agreed to resubmit its response to RAI 650.004, in light of the staff's concern. Pending an acceptable resolution of this concern, the staff considers the capability of the AP1000 IRWST screens to accommodate anticipated debris loadings to be DSER Open Item 6.2.1.8.2-1.

Westinghouse Response:

Westinghouse revised its response to RAI 650.004 in order to address the NRC concerns as discussed in our teleconference on April 3, 2003. The revised RAI response was submitted to the NRC on April 24, 2003 in letter DCP/NRC1580. Based on discussions with the NRC after the issuance of the DSER, it was agreed that this response satisfactorily addressed the NRC concerns except for the calculated pressure loss. Westinghouse agreed to revise the calculation of the pressure loss across the IRWST screens. The revised calculation is based on the following:

1. There are 500 lb of resident debris (fiber and particles) located inside containment,
2. This debris is assumed to be neutrally buoyant (both fibers and particles) such that they are easily transported with flow.
3. The resident debris is distributed around the containment in proportion to the floor areas.
4. If a floor area sees flow either from LOCA blowdown, ADS venting or containment recirculation, then debris associated with that floor area is transported to a screen.
5. If a floor area does not see flow (whether it floods or not) then none of the debris assigned to that floor area is assumed to be transported.

6. The head losses across the screens will be calculated using the BLOCKAGE code. The resident debris fiber material is assumed to be represented by NUKON.
7. Sensitivity studies will be performed with variations in both the amount of debris transported to the screens and in the mass ratio of fiber versus particulate debris.

Based on these assumptions, methods and approaches, the pressure drop analysis performed for RAI 650.004 (IRWST screens) was revised.

Ability of IRWST Screens to Tolerate Debris

Even though there is a low probability of having debris in the IRWST and having that debris transported to the screens in the IRWST, the IRWST screens and the PXS have significant capability to tolerate debris. A bounding analysis of the pressure drop that could be caused by debris (fiber and particle) on the IRWST screens has been performed for the AP1000.

The assumptions used in the analysis include:

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

- All of the debris is neutrally buoyant (fiber and particle) such that it is easily transported by flow. No credit is taken for settling or trapping of debris other than on the screens.

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

- The limiting break location with respect to debris loading on the IRWST screens has been determined to be the break of a pipe connected to the top of the Pressurizer that washes a portion (33%) of the operating deck into the IRWST via the gutter. This event results in less than 260 lb (of the 500 lb) of debris being transported to PXS screens.

- The debris deposited on any screen is assumed to be based on the flow split about containment. As noted above, for the LOCA of a pipe connected to the top of the Pressurizer, a total of about 260 lb resident containment debris is available for transport. Of this amount of debris, about 130 lb is transported to the IRWST screens. The remainder (130 lb) is transported to the containment recirculation screens.

- There are two screens in the IRWST, each with an area of 70 ft2. With the limiting break location, there is injection from both IRWST lines into the RCS. Note that even with a DVI break there is flow through both IRWST lines although one spills.

- The plant response to the break of a pipe at the top of the Pzr will be similar to its response to a spurious ADS LOCA. In a spurious ADS, the flow rate through each IRWST screen is

90 lb/sec (DCD figure 15.6.5.4B-13, -14). Note that for a DVI LOCA the IRWST flows are greater, as much as 160 lb/sec (at about 2700 sec, DCD figure 15.6.5.4B-71). In order to bound the IRWST flows and screen debris loadings, the DVI LOCA flows are used in this analysis. At the IRWST water conditions for this event, the 160 lb/sec mass flow results in a volumetric flow rate of 1170 gpm.

- At this flow rate, the screen face velocity is 0.037 ft/sec.

- With the above amounts of debris and flow rates, the pressure loss across the debris is calculated by the BLOCKAGE code to be less than 0.03 psi. A summary of BLOCKAGE Code input and resulting output for the base case are shown on the table that follows. Refer to DSER OI response to 2.1.8.3-3 for additional discussion on the use of the BLOCKAGE code.

- In addition to the base calculation, sensitivity calculations were performed on the amount of debris transported and the mass ratio of fiber to particulate debris.
  - Sensitivity calculations were performed varying the total mass of material from 80% (104lb.) to 120% (156lb.) of the base case (130 lb.) This sensitivity addresses possible variability in the amount of debris available to transport.
  - Fiber to particulate mass ratios ranging from 30% fiber/ 70% particulate to 70% fiber/ 30% particulate were investigated for all three total mass cases. This sensitivity addresses the impact of fiber to particulate ratios different from the base case assumption of a 50/50 split.

  The results of these sensitivity analyses are shown on the table that follows. The pressure drops across the screens for all the cases investigated ranged from 0.02 psid to 0.04 psid. For the range of masses and mass ratios investigated, the range of calculated pressure drop values was narrow and the trend of pressure drops within the range showed no unexpected results.

- The pressure loss through the intact IRWST injection line from the IRWST to the RV downcomer is more than 5.8 psi at this flow rate. The increase in screen DP for the base case shown above is only 0.03 psi or less than 4%. The IRWST injection flow would only have to decrease an insignificant amount to compensate for this increase in screen DP. The potential impact on injection flow is insignificant.

In summary, it is concluded that the current AP1000 design is not susceptible to degradation of IRWST gravity injection flow due to IRWST screen blockage resulting from deposition of resident containment debris on the screens.

IRWST SCREEN

	Mass Debris (lbm)	Percent Fiber	Percent Particulate	Mass Fiber (lbm)	Mass Particulate (lbm)	Thickness (in)	Pressure Drop (ft-water)	Pressure Drop (psi)	Blockage Case Name
100% of Total Debris	130								
	130	30%	70%	39	91	1.39	0.05	0.02	APRS_201
	130	40%	60%	52	78	1.86	0.05	0.02	APRS_202
Base Case	130	50%	50%	65	65	2.32	0.06	0.03	APRS_203
	130	60%	40%	78	52	2.78	0.07	0.03	APRS_204
	130	70%	30%	91	39	3.25	0.08	0.03	APRS_205
80% of Total Debris	104	30%	70%	31	73	1.11	0.04	0.02	APRS_211
	104	40%	60%	42	62	1.49	0.04	0.02	APRS_212
	104	50%	50%	52	52	1.85	0.05	0.02	APRS_213
	104	60%	40%	62	42	2.23	0.06	0.03	APRS_214
	104	70%	30%	73	31	2.60	0.07	0.03	APRS_215
120% of Total Debris	156	30%	70%	47	109	1.67	0.06	0.03	APRS_206
	156	40%	60%	62	94	2.23	0.06	0.03	APRS_207
	156	50%	50%	78	78	2.78	0.07	0.03	APRS_208
	156	60%	40%	94	62	3.34	0.09	0.04	APRS_209
	156	70%	30%	109	47	3.90	0.10	0.04	APRS_210

INPUT TO BLOCKAGE CODE

Value	Parameter	Description	Note
AP1000 Calculation Fiber and Particulate Debris Parameters			
0.986	ε_f	Pure fiber bed porosity	NUKON, Reference 2
175	ρ_f	Fiber density (lb_m/ft^3) also Material density	NUKON, Reference 2
2.4	c_0	Fabricated Fiber Density	NUKON, Reference 2
68.64	ρ_p	Particle density (lb_m/ft^3)	Specific Gravity of 1.1
1.71E+05	Sv	Specific (volumetric) surface area (ft^2/ft^3)	NUKON, Reference 2
AP1000 IRWST Screen			
1170		Flow of Water though Recirc. Screen	
120		Temperature of water at screen (deg. F.)	
140		IRWST screen (2 x 70 ft2) total flow area (ft^2)	
130		Mass of Total Debris (lbm) Base Case	

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

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

Original RAI Number(s): 650.001

Summary of Issue:

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

Westinghouse Response:

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

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

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

Resident Fibrous and Particle Debris:

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

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

- A total of 500 lb of resident debris is located in the containment (DSER OI 2.1.8.3-3 R1). The base case assumes that this debris is divided 50/50 between fibers and particles. Also, as described below, sensitivity studies are also performed assuming a range of particulate to fiber ratios.
- The debris is distributed around the containment in proportion to the floor areas. As discussed in DSER OI 6.2.1.8.3-3 R1, not all of this debris will be transported because some floor areas will not see flow during a LOCA.
- The limiting break location with respect to maximizing the debris that might enter the RCS has been determined to be a DVI break in a loop compartment. Such a break will result in none of the operating deck and only a portion of the CMT room floor (< 67%) seeing flow. As a result, less than 250 lb of resident debris will be transported.
- The debris deposited on any screen is assumed to be based on the flow distribution in the containment. As noted above, for the DVI break in a loop compartment, less than 250 lb of resident containment debris is available for transport. Of this amount of debris, about 100 lb of debris will be transported to the IRWST screens. The remainder (150 lb) will be transported to the recirculation screens and to the RCS via the break. This 150 lb is further divided in the proportion of the relative flows as described below.
- Conservative analyses have shown that 60% of the total flow to the core is through the break and 40% through the recirculation screens. Assuming the debris transport is proportional to the flow, 60% of the resident debris will enter the RCS through the break (90 lb). The other 40% (60 lb) would be trapped on the two recirculation screens. These debris amounts are based on the relative flows through the break and through the PXS recirc lines as shown on DCD figures 15.6.5.4C-13 and -14 after 7000 sec. Although the flow through the break into the RCS starts earlier than through the PXS recirc lines, it

would take many hours to transport all of the debris to the RCS / recirc screens. For example, the total water mass in the containment floodup areas is about 5,236,000 lb. At a recirc flow of 180 lb/sec it would take about 10 hours for all of this water to flow through the RCS. The situation for the recirc screens is much less limiting than that discussed in DSER OI response 2.1.8.3-3 R1.

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

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

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

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

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

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

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

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

In summary, the bounding pressure loss through resident debris that might deposit on the lower core support plate or in the core would not reduce the flow to the core.

Core Pressure Drop

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

INPUT TO BLOCKAGE CODE

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

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Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 6.2.1.8.3-3 (Response Revision 1)

Original RAI Number(s): 650.005

Summary of Issue:

The staff's review found that insufficient information was available in the DCD to determine whether the containment recirculation screens are capable of tolerating anticipated post-accident debris loadings. Therefore, in RAI 650.005, the staff requested additional information from the applicant to determine the debris-blockage failure criterion of the containment recirculation screens. The applicant responded to RAI 650.005 in a letter dated February 21, 2003, by providing an analysis intended to demonstrate that the AP1000 recirculation screens could accommodate a mass of resident debris (i.e., 227 kg, or 500 lb) that is equivalent to estimates made for current generation PWRs in the GSI-191 parametric study (NUREG/CR-6772). However, the staff could not accept this analysis, primarily because the applicant assumed that a single density value is valid for all density-dependent calculations regarding resident fibrous debris. According to the physical properties of analyzed types of fibrous materials, potentially different density values may be required to correctly determine the settling velocity (i.e., the material density), to calculate a volume from the assumed mass (i.e., the "as-found" density), and to determine the thickness and porosity of the associated debris bed (i.e., the rubblized density). As a result of the applicant's single-density assumption, which deviated significantly from the material properties of the low-density fiberglass on which the head loss data referenced by the applicant was based, the NRC staff concluded that the calculation was unacceptable. During a teleconference on April 3, 2003, the applicant agreed to resubmit its response to RAI 650.005, in light of the staff's concern. Pending an acceptable resolution of this concern, the staff considers the capability of the AP1000 containment recirculation screens to accommodate anticipated debris loadings to be DSER Open Item 6.2.1.8.3-3.

Westinghouse Response:

Westinghouse revised its response to RAI 650.005 in order to address the NRC concerns as discussed in our teleconference on April 3, 2003. The revised RAI response was submitted to the NRC on April 24, 2003 in letter DCP/NRC1580. Based on discussions with the NRC after the issuance of the DSER, Westinghouse agreed to revise the calculation of the pressure loss across the containment recirculation screens. The revised calculation is based on the following:

1. There are 500 lb of resident debris (fiber and particles) iassumed to be located inside containment.
2. This debris is assumed to be neutrally buoyant (both fibers and particles) such that they are easily transported with flow.
3. The resident debris is distributed around the containment in proportion to the floor areas.
4. If a floor area sees flow either from LOCA blowdown, ADS venting or containment recirculation, then debris assigned to that floor area is assumed to be transported to a screen.

5. If a floor area does not see flow (whether it floods or not) then none of the debris assigned to that floor area is assumed to be transported.
6. The head losses across the screens will be calculated using the "BLOCKAGE" code. The resident debris fiber material is assumed to be represented by NUKON.
7. Sensitivity studies are performed with variations in both the amount of debris transported to the screens and in the mass ratio of fiber versus particle debris.

Based on these assumptions, methods and approaches, the head loss analysis performed for RAI 650.005 (containment recirculation screen) was revised.

Ability of Containment Recirculation Screens to Tolerate Debris

Fibrous insulation is not used inside containment of the AP1000 where it can be damaged by LOCA blowdown jets and is therefore not considered in responding to this item.

A potential source of debris is resident fiber and particles inside containment. Such debris might be close enough to the density of water that it would stay suspended in the containment water long enough that it could be transported to containment recirculation screens and possibly also into the RCS through a break that becomes flooded. Thus, resident containment debris will be used for calculations performed to respond to this DSER OI.

The parametric sump blockage evaluation performed for GSI-191 and reported in LA-UR-01-4083 (Reference 1) assumed a range of 100 lb to 500 lb for latent containment debris. The larger value will be used for this evaluation even though the AP1000 containment is smaller than a large dry containment used for a typical 4-loop plant. Further, it will be assumed for the base case that 50% or one half of the latent containment debris will be in the form of fiber. Sensitivity studies will be performed to show the impact of varying this distribution.

An evaluation was performed to determine the amount of the resident debris that might be transported to the recirc screen. This evaluation assumed that:

- The resident debris is distributed around the containment in proportion to the floor areas.

- If a floor area sees flow from LOCA blowdown, ADS venting or containment recirculation, then debris associated with that floor area is assumed to be transported to a screen.

- If a floor area does not see flow (whether it floods or not) then none of the debris assigned to that floor area is assumed to be transported.

In the AP1000, there are two significant floor areas that do not usually see flow following a LOCA. These areas include the operating deck and the CMT floor. Since the AP1000 does not have a automatic spray system that will be used during design basis accidents, the operating deck will not see flow from a spray system as in current PWRs. The AP1000 does have a nonsafety-related spray feature that will only be used during severe accidents; the feature

requires local manual actions. The CMT floor will not usually see flow from a LOCA unless the break is a DVI or CMT inlet break.

For the containment recirculation screens, the limiting case is a DVI break in a PXS room. This break location does not cause flow on the operating deck because of the arrangement of the AP1000 containment. It is assumed that a DVI break can cause flow on a portion of the CMT floor during the LOCA blowdown. This flow would exit the PXS room through vents in its roof (CMT floor) and would allow spray / overflow from the room to wash a portion of the CMT floor (< 50%). Note that there are conservatisms in this calculation including no credit for settling and no credit for debris being flushed down into the reactor vessel cavity by the initial LOCA blowdown flow. As a result of this evaluation, it is calculated that less than 250 lb (of the 500 lb in the containment) will be transported.

The postulated DVI line break in a PXS room is also limiting for the recirculation screens because it is assumed to render the recirculation valves located in the PXS room inoperable since they will be flooded before they are actuated. As a result, all of the recirculation flow will pass through a single recirculation screen. This condition maximizes the amount of debris transported to and collected on the single operating recirculation screen, and head loss across the resulting debris bed. In addition, a DVI break in one of the PXS rooms results in lower containment flood levels. For LOCAs in other locations, both recirculation screens will be available which will result in lower flow rates through each screen. In addition, the containment flood levels will be higher providing a greater driving head for injection.

For the DVI break in a PXS room, about 150 lb of resident debris will be transported to the one operable containment recirculation screen (of the 250 lb of debris transported). The remainder (100 lb) will be transported to the IRWST screens.

The volume of latent fibrous debris inside containment is calculated as:

$$\textit{debris volume} = \textit{debris mass} / \textit{debris density}$$

Using the characteristics for NUKON from NUREG/CR-6224 as the fiber in our analysis :

$$\textit{fabricated debris density} = 2.4\ lb/ft^3$$

Using this fabricated density (2.4 lb/ft3) and the fibrous debris mass (50% of 150 lb debris or 75 lb.) that reaches the recirculation screen, the volume of fibrous debris is calculated to be:

$$Volume = 31.2\ ft^3$$

A single AP1000 recirculation screen has a flow area of 70 ft2. Thus, assuming that all the fibrous debris is deposited onto the one operating recirculation screen, the thickness of the debris bed is calculated to be:

$$thickness = 31.2 ft^3 /(70\ ft^2) = 0.446\ ft = 5.35\ inches$$

NUKON was selected as the fiber for this evaluation. From previous work, it is known that saturated NUKON will be buoyant and thus will likely be transported to the screen. In addition the characteristics of NUKON are well documented in NUREG/CR-6224 and the correlation in the BLOCKAGE Code is verified for NUKON fibers.

It is recognized that even at low flow rates, buoyant-neutral debris will tend to be transported to the recirculation screen. Although from Table B-3, "Fibrous Debris Classification by Shape," of NUREG/CR 6224 (Reference 2), it is noted that even single strands of fiber will settle in calm pools. This evaluation took no credit for settling of fibrous debris.

Under long term cooling for a postulated DVI line break, the flow rate through the single operating recirculation screen is estimated to be about 180 lb/sec (refer to DCD figures 15.6.5.4C-13 and -14). With expected containment recirculation water temperatures, the flow through the recirc screen would be 1335 gpm.

The pressure drop through the mixed fiber bed is calculated using the BLOCKAGE Code, documented in NUREG/CR-6371, Reference 3, which uses the method described in Appendix B of Reference 2. Several assumptions used in the calculation are identified below:

- The fiber characteristics used in the analysis are the same as NUKON (fiberglass) in Reference 2. Due to the high porosity (98.6%), the use of NUKON fiber characteristics resulted in a relatively thick fibrous bed. This is taken to be a reasonable approximation.
- The specific gravity of the particulate debris is taken as 1.1. Assuming a low specific gravity maximizes the volume of particles and therefore the impact on calculated pressure drop.
- The cases run in BLOCKAGE were established in the "User Specified Volumes" mode. That is, the volume of material was iterated until the code specified mass of material deposited on the screen matched the mass for the case being run. This is necessary since the code iterates the deposited volumes based upon the concentration of material in the pool.

Using a flow of 1335 GPM relating to a velocity of 0.04 ft/sec and the BLOCKAGE 2.5 Code, the head loss through a mixed fiber bed on the recirculation screen is calculated to be:

$$\Delta P = 0.18 \text{ psid}$$

In addition to the base calculation, sensitivity studies were performed on the amount of debris transported and the ratio of fibers / particles.

- Sensitivity calculations were performed varying the total mass of material from 80% (120 lb.) to 120% (180 lb.) of the base case (150 lb). This sensitivity addresses possible variability in the amount of debris available to transport.

- Fiber to particulate mass ratios ranging from 30% fiber/ 70% particulate to 70% fiber/ 30% particulate were investigated for all three total mass cases. This sensitivity addresses the impact of fiber to particulate ratios different from the base assumption of a 50/50 split.

The results of these sensitivity analyses are shown on the table that follows. The pressure drops for all the cases investigated ranged from 0.11 psid to 0.29 psid. For the range of masses and mass ratios investigated, the range of calculated pressure drop values was narrow and the trend of pressure drops within the range showed no unexpected results.

The assumption of a 50 / 50 split between latent fibrous and particulate debris is considered a reasonable engineering judgement. In addition, reasonable variations of this split (30/70 to 70/30) only have second order effects on the calculated DP.

The base calculated increase in pressure drop through the recirculation screen, assuming the collection of resident containment debris on the screens, is small (~6%) compared with the 2.8 psi in these lines during the DCD analysis. Note that the recirculation flow would only have to decrease ~ 3% to compensate for this increase in screen DP. Also note that if best estimate PXS line resistances were assumed, it would completely compensate for the debris DP. The potential impact on PXS flow, especially with best estimate line resistances, is insignificant.

The above calculated increase in pressure drop due to a mixed fiber-particulate is considered conservative for the following reasons:

- The limiting flow case was assumed. That is, only one of the two recirculation screens was taken to be operable due to the assumed break location. This provided for a maximum velocity to and through the operating recirculation screen, also maximizing the potential for debris transport to the operating recirculation screen.

- The value of 500 lb of resident containment debris inside containment is based on a value that was scaled from BWR's to be representative of a large dry containment for PWR's. The floor area of the AP1000 is less than that of a current large dry containment for PWR's. Thus, the initial mass of resident containment debris used for the AP1000 evaluation is conservative.

- The total amount of latent containment debris used in the evaluation is considered large. An aggressive foreign materials exclusion program and good housekeeping practices are expected to maintain latent containment debris sources well below the 500 lb level.

- A conservative estimate of the maximum debris loading on the containment recirculation screen is assumed. Debris on floor areas subject to flow from the event is assumed to reach the screens. No credit is taken for settling of this debris.

In summary, it is concluded that the current AP1000 design is not susceptible to loss of natural circulation of coolant from the containment due to recirculation screen blockage resulting from deposition of latent containment debris on the recirculation screen.

References:

1. LA-UR-01-4083, Revision 1, "GSI-191: Parametric Evaluations for Pressurized Water Reactor Circulation Sump Performance," dated August 2001

2. Regulatory Guide 6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," dated August 1994

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

RECIRCULATION SCREEN									
	Mass Debris (lbm)	Percent Fiber	Percent Particulate	Mass Fiber (lbm)	Mass Particulate (lbm)	Thickness (in)	Pressure Drop (ft-water)	Pressure Drop (psi)	Blockage Case Name
100% of Total Debris	150								
	150	30%	70%	45	105	3.21	0.33	0.14	APRS_101
	150	40%	60%	60	90	4.29	0.37	0.16	APRS_102
Base Case	150	50%	50%	75	75	5.35	0.42	0.18	APRS_103
	150	60%	40%	90	60	6.42	0.48	0.21	APRS_104
	150	70%	30%	105	45	7.53	0.57	0.25	APRS_105
80% of Total Debris	120	30%	70%	36	84	2.57	0.26	0.11	APRS_111
	120	40%	60%	48	72	3.43	0.29	0.13	APRS_112
	120	50%	50%	60	60	4.30	0.34	0.15	APRS_113
	120	60%	40%	72	48	5.14	0.39	0.17	APRS_114
	120	70%	30%	84	36	6.00	0.45	0.20	APRS_115
120% of Total Debris	180	30%	70%	54	126	3.85	0.39	0.17	APRS_106
	180	40%	60%	72	108	5.14	0.44	0.19	APRS_107
	180	50%	50%	90	90	6.43	0.50	0.22	APRS_108
	180	60%	40%	108	72	7.71	0.58	0.25	APRS_109
	180	70%	30%	126	54	9.00	0.68	0.29	APRS_110

		INPUT TO BLOCKAGE CODE	
Value	Parameter	Description	Note
AP1000 Calculation Fiber and Particulate Debris Parameters			
0.986	ε_f	Pure fiber bed porosity	NUKON, Reference 2
175	ρ_f	Fiber density (lb_m/ft^3) also Material density	NUKON, Reference 2
2.4	c_0	Fabricated Fiber Density	NUKON, Reference 2
68.64	ρ_p	Particle density (lb_m/ft^3)	Specific Gravity of 1.1
1.71E+05	Sv	Specific (volumetric) surface area (ft^2/ft^3)	NUKON, Reference 2
AP1000 Recirculation Screen			
1335		Flow of Water though Recirc. Screen	
175		Temperature of water at screen (deg. F.)	
70		Single recirculation screen flow area (ft^2)	
150		Mass of Total Debris (lbm) Base Case	

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 9.5.1-2 Response Revision 1

Original RAI Number(s): 280.011

Summary of Issue:

In RAI 280.011, the NRC staff raised a concern that 41percent of the total fire induced core damage frequency (CDF) is assigned to containment. The containment fire is such a large contributor, and there are areas in containment which exist where redundant safe shutdown components required following a fire have not been separated by complete fire barriers. Therefore, the NRC staff requested that the applicant perform a mathematical fire model in accordance with NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." The fire model should demonstrate that a fire would be confined to the zone of origin such that redundant components remain free of fire damage. The applicant selected the fire-induced vulnerability evaluation (FIVE) methodology (EPRI TR-100370, "Fire-Induced Vulnerability Evaluation," issued April 1992), which is not a mathematical fire model. FIVE was approved by the NRC in the early 1990's primarily as a tool to provide a qualitative assessment of fire risk for the individual plant examination of external events (IPEEE) to perform fire probabilistic risk assessments (PRAs). The FIVE methodology is limited in that large open areas, such as those in containment, are not capable of being realistically modeled. Therefore, the NRC staff expressed concern that the FIVE methodology was not appropriate to model fires within containment.

The applicant responded to the RAI and stated that NFPA 805 permits the use of the FIVE methodology. The staff responded that Appendix C Section C.2.2., "Fire Model Features and Limitation" of NFPA 805 specifically states that the limitations of each fire model should be taken into consideration, in order to produce reliable results, that will be useful in decision making. This section specifically states that *"Some models may not be appropriate for certain conditions and can produce erroneous results if applied incorrectly."* The intent of the Appendix C, Table C.2.2.(b), is to enable the user to select the appropriate model for a particular fire area, in order to obtain useful estimates to best approximate the conditions within an enclosure as a result of an internal fire. In addition, NFPA 805 states that the fire model shall be acceptable by the authority having jurisdiction (AHJ). In this case, the AHJ is the NRC. The use of the FIVE methodology has not been accepted outside of the IPEEEs at the NRC. The staff does not agree that the use of FIVE is an appropriate choice to model a fire within containment. Therefore, this item is unresolved.

Westinghouse Response:

Westinghouse believes that our licensing submittals related to fire protection have satisfied the written regulatory requirements and guidance for Design Certification. Westinghouse has provided a fire hazards analysis in DCD Appendix 9A that demonstrates that the A1000 complies with or requests exemptions from the requirements BTP CMEB 9.5-1. AP1000 has used a deterministic-based approach for the fire evaluation described in Chapter 9 and

Appendix 9A of the AP1000 Design Control Document (DCD). Consistent with Chapter 1 and Figure 2.2 (Methodology) of NFPA 805, AP1000 has chosen this deterministic approach to justify its compliance with the fire protection requirements of 10CFR50 and 10CFR52. NFPA 805 clearly indicates that the designer may use *either* the deterministic *or* the probabilistic method for fire evaluation. Since this fire hazards evaluation was deterministic, it "involves implied, but unquantified, elements of probability in the selection of specific accidents to be analyzed as design basis events" (see NFPA 805, Section 1.6.11). The FIVE methodology was not used in the fire hazards analysis documented in AP1000 Appendix 9A to justify compliance with regulatory fire protection requirements.

In addition, to comply with an NRC request to have an AP1000 specific PRA, Westinghouse has performed a fire PRA as part of the overall AP1000 PRA. 10 CFR Part 52 requires an applicant to submit a plant-specific PRA, although it does not specifically require a fire PRA. The results of the fire PRA have shown that AP1000 plant risk due to a fire is extremely low. The AP1000 PRA methodology included using the FIVE methodology inside containment at NRC's request. The FIVE methodology is an acceptable methodology for probabilistic analysis in accordance with NFPA 805. The Revision 1 response to RAI 280.011 and the current revision of the AP1000 Probabilistic Risk Assessment report describes the method used for AP1000. They also describe that the method used may be overly conservative and that additional safety-related function or component based assessments were performed to ensure the design has a very low risk from fires in containment affecting its safety. Although fires in containment represent the largest percentage contributor to CDF, the overall CDF itself is acceptably small. No further refinement of the PRA is needed. In many areas of the PRA, more detailed or sophisticated analysis may lead to improvements in the PRA results. It is acceptable to conservatively simplify the PRA analysis.

Westinghouse concludes that AP1000 has met the applicable fire-related regulations, and that no additional fire analysis is required.

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

Original RAI Number(s): 440.169

Summary of Issue:

The applicant's submittals did not provide sufficient justification that the models and correlations in NOTRUMP or WCOBRIVTRAC have been adequately assessed to cover the ranges expected to occur in the upper plenum of the API 000. While correlations exist to model upper plenum entrainment phenomena, the issue that remains is adequacy of the database. Existing correlations are based on relatively small diameter vessels, low gas flow rates, and for some data, air-water as opposed to steam-water. Because of the small vessel size in these data, conditions were essentially one-dimensional. Flow in the upper plenum of the AP1000 is expected to be non-uniform and three dimensional. Thus, a suitable database for assessing entrainment correlations in the upper plenum has not been established, Given the lack of well scaled experimental data on upper plenum entrainment phenomena and the importance of predicting this process in an advanced plant SBLOCA transient, it is recommended that new experimental data be obtained to support the use of the upper plenum entrainment models in the AP1000. This data was requested by the NRC staff in a letter dated March 18, 2008, from J, Lyons. Therefore, this is DSER Open Item 21.5-2.

Westinghouse Response:

As indicated in Westinghouse letter DCP/NRC1604 dated July 14, 2003, additional testing has been performed in the Oregon State University APEX1000 test facility that is specifically scaled to the AP1000. The test facility description report, the facility scaling report relative to AP1000, and test summary reports for several tests were previously submitted to NRC (Westinghouse letter DCP/NRC1595 dated June 2, 2003) in support of the AP1000 design certification review.

Two APEX1000 tests are discussed here. They are labeled as tests DBA02 and DBA03. Both of these tests simulate an AP1000 double ended direct vessel injection line (DEDVI) break. Test DBA02 simulates an ADS4 failure on the non-pressurizer side of the plant, and DBA03 simulates an ADS4 failure on the pressurizer side of the plant. These tests can be used to examine the effect of the AP1000 power level on system behavior and core and upper plenum inventory in particular.

Figure 1 shows the wide range downcomer pressure for DBA-02 and DBA-03 double-ended DVI break tests. For both tests, the pressure quickly falls due to the size of the break, then the rate of decrease slows as the reactor coolant system becomes saturated near the saturation pressure of the steam generator secondary side. The automatic depressurization system is actuated on the CMT level on the broken side with ADS-1 actuation at about 90 seconds. At this time, the pressure falls more rapidly as ADS-2 and ADS-3 are actuated in sequence. At about 250 seconds, the ASD-4 is actuated and the downcomer pressure falls. Figure 2 shows the narrow range downcomer pressure. The pressure is higher for DBA-02 due to the

interaction between the pressurizer and flow out the ADS-4 valves. The assumed single failure for DBA-02 is on the non-pressurizer side ADS-4. This results in 100% ADS-4 flow area on the pressurizer side, and 50% flow area on the non-pressurizer side. When the larger ADS-4 path is on the pressurizer side, the increased steam flow to that side causes the pressurizer to fill more and then drain more slowly (Figure 3) relative to having the larger ADS-4 path on the opposite hot leg. Figure 4 shows the total ADS-4 liquid flow. The effect of ADS-4 failure location is to reduce the ADS-4 liquid flow for a period of time during DBA-02.

The total DVI injection flow is shown in Figure 5. The flow from the CMTs for both tests stops at 900 seconds when the CMTs are empty. For DBA-03, the IRWST injection occurs before the end of CMT injection, but for DBA-02, an injection gap of about 300 seconds occurs before IRWST injection is established.

A capacitance probe was installed for DBA-02 and DBA-03. This probe measures the two-phase liquid level in the upper plenum. Figure 6 shows the two-phase level for these tests. In both cases, the level falls to the hot leg elevation and remains there until ADS-4 actuation. At that time, the level drops slightly into the upper plenum well above the top of the core, then recovers to the hot leg elevation. For DBA-02, during the injection gap, the mixture level falls slightly below the bottom of the hot leg, but remains significantly higher than the top of the core.

Figures 7 and 8 show the collapsed liquid level and the average void fraction in the core region. The collapsed level drops after ADS-1 is actuated, and the minimum collapsed level occurs after ADS-4 actuation. The average void fraction in the core is nearly 70% at this time. The core inventory recovers steadily as the accumulator and CMT inject. For DBA-02, the gap between the end of CMT injection and the start of IRWST injection results in a reduction in the core inventory, reaching 50% core-average void fraction just before IRWST injection is established.

Figure 9 shows the peak heater rod temperature. The temperature is measured inside the rod which accounts for the temperature elevation above saturation. For both tests, the temperature follows the decay power curve and there is no heatup, indicating the heater rods are adequately cooled.

Figures 10 through 12 show the test results during the long term cooling phase of the tests. Figure 10 shows the capacitance probe two-phase level during the long-term cooling phase of the test (i.e. IRWST-sump injection). In both cases, the level recovers to the top of the hot leg and remains there until the end of the test, and is significantly higher than the top of the core.

Figure 11 shows the collapsed liquid level in the core region for the long term cooling phase. After stable IRWST and sump injection are established, the collapsed level stabilizes near the top of the core. The level falls slightly in DBA-02 due to a mismatch in the levels in the IRWST and sump at the time of sump injection.

Figure 12 shows the peak heater rod temperature. For both tests, the temperature follows the decay power curve and there is no heatup, indicating the heater rods are adequately cooled.

The APEX-1000 DEDVI tests show the following:

1. Two-phase mixture level remains in the upper plenum at all times for both design basis tests.
2. Core-average void fraction reaches a maximum of 70% just after ADS-4 actuation, then decreases to about 30%. For DBA-02, the injection gap before IRWST injection causes an increase in the void fraction to 50%.
3. The AP1000 ADS4 flow capacity as simulated in these tests is sufficient to draw liquid flow through the core and into the upper plenum throughout the transient. This system behavior is the same as seen in integral tests scaled to AP600 and indicates the system behavior is not sensitive to upper plenum entrainment phenomena within the range of AP600 and AP1000 conditions. This is consistent with the insensitivity to upper plenum and hot leg entrainment observed in the sensitivity study discussed in the response to Open Item 21.5-1.
4. The peak heater rod temperature show no temperature excursions which shows that there is effective two-phase heat transfer at the heater rods at all times.

These tests and the other design basis accident tests performed at the APEX-1000 test facility show that the effect of upper plenum entrainment on the passive core cooling system's ability to assure core coolability is not significant. For each test, the two-phase level always remains in the upper plenum, and the core remains covered for all phases of the simulated accident.

Since the APEX-1000 facility is well scaled to AP1000, these tests form the body of data requested in the NRC staff letter dated March 18, 2003 from J. Lyons. This data will be used as the basis for the verification and validation of NOTRUMP and WCOBRA/TRAC will be provided in a separate transmittal and will be included in an update to WCAP-15644, AP1000 Code Applicability Report.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

a, b, c

Figure 1: Downcomer Pressure – Wide Range

a, b, c

Figure 2: Downcomer Pressure - Narrow Range

a, b, c

Figure 3: Pressurizer Mass

a, b, c

Figure 4: Integrated ADS-4 Liquid Flow

a, b, c

Figure 5: Total DVI Injection Flow

a, b, c

Figure 6: Two-Phase Mixture Level in Upper Plenum

a, b, c

Figure 7: Core Collapsed Liquid Level

a, b, c

Figure 8: Core Average Void Fraction

a, b, c

Figure 9: Peak Heater Rod Temperature

a, b, c

Figure 10: Two-Phase Mixture Level In Upper Plenum - Long Term Cooling

a, b, c

Figure 11: Core Collapsed Liquid Level – Long Term Cooling

a, b, c

Figure 12: Peak Heater Rod Temperature – Long Term Cooling

DSER Open Item Number: 21.5-2 (Response Addendum 1)

Original RAI Number(s): 440.169

Summary of Issue:

The applicant's submittals did not provide sufficient justification that the models and correlations in NOTRUMP or WCOBRA/TRAC have been adequately assessed to cover the ranges expected to occur in the upper plenum of the API 000. While correlations exist to model upper plenum entrainment phenomena, the issue that remains is adequacy of the database. Existing correlations are based on relatively small diameter vessels, low gas flow rates, and for some data, air-water as opposed to steam-water. Because of the small vessel size in these data, conditions were essentially one-dimensional. Flow in the upper plenum of the AP1000 is expected to be non-uniform and three-dimensional. Thus, a suitable database for assessing entrainment correlations in the upper plenum has not been established, Given the lack of well scaled experimental data on upper plenum entrainment phenomena and the importance of predicting this process in an advanced plant SBLOCA transient, it is recommended that new experimental data be obtained to support the use of the upper plenum entrainment models in the AP1000. This data was requested by the NRC staff in a letter dated March 18, 2008, from J, Lyons. Therefore, this is DSER Open Item 21.5-2.

Westinghouse Response: (Addendum 1)

Westinghouse has provided a previous response to DSER Open Item 21.5-2. This response provides additional information to resolve this DSER Open Item and is labeled 21.5-2P Addendum 1.

In order to provide additional justification for the applicability of the NOTRUMP computer code to the AP1000 plant design, a sensitivity study was performed with the NOTRUMP model for AP1000 which increased upper plenum and hot leg entrainment as described in Reference 21.5-1.1. Additionally, a test plan was developed at the Oregon State University APEX Test facility as modified to reflect the AP1000 plant design. The detailed description of the APEX Test facility, as modified to reflect the AP1000 design, can be found in Reference 21.5-1.2. The APEX test facility has been scaled to the AP1000 plant design, as described in Reference 21.5-1.3, to assure appropriate facility response and support further computer code validation.

The results of the Reference 21.5-1.1 sensitivity study indicate that the AP1000 behavior is relatively insensitive to the amount of entrainment from the upper plenum and hot legs. The Reference 21.5-1.1 sensitivity study was performed assuming that the upper plenum, hot-leg, PRHR inlet and ADS-4 fluid nodes were homogenous, at the time of ADS-4 actuation, which increases potential liquid entrainment. The results indicate sufficient inventory remains in the vessel such that adequate core cooling is maintained. Reference 21.5-1.1 can be reviewed for additional details on the results of this study.

To further confirm the applicability of the NOTRUMP computer code to predict the AP1000 plant behavior for Small Break Loss Of Coolant Accidents (SBLOCAs), the revised OSU APEX test facility (References 21.5-1.2 and 21.5-1.3) was modeled with the Advanced Plant version of the NOTRUMP computer code. The noding diagram utilized for the Reference 21.5-1.3 OSU APEX simulations can be found in Figure 21.5-2.1. The model utilized for these simulations is similar to that utilized for the AP600 APEX simulations with the following exceptions:

- Revised noding in the Pressurizer
- Revised noding in the Core Makeup Tanks

The Pressurizer noding was altered from a single fluid node to multiple fluid nodes (See Figure 21.5-2.2) for several reasons. Firstly, the APEX facility was modified to accommodate the increase in Pressurizer volume required to represent the AP1000 plant design. The modification was such that a section was added to the upper pressurizer. This upper section is a larger diameter than the lower section. Therefore, to properly model the change in geometry requires an additional fluid node be added to the NOTRUMP model. In addition, to improve the predicted void distribution in the Pressurizer, additional fluid nodes were added to represent the Pressurizer surge line and split the common fluid node section, representing the Pressurizer tank, into [ ]^{a,c} individual fluid nodes as can be seen in Figure 21.5-2.2.

The Core Makeup Tank (CMT) model was revised to add additional fluid nodes to enhance the fluid temperature distribution predicted by the NOTRUMP code. Since the NOTRUMP code does not have a thermal stratification model, when warm fluid is introduced to a fluid node, it is assumed to perfectly mix with the existing fluid node. As such, when only a few fluid nodes are modeled, the fluid temperature at the bottom of the CMT begins to artificially heat due to the numerical mixing effect. A sensitivity study was performed which altered the CMT noding from the standard [ ]^{a,c} model to a [ ]^{a,c} model in the AP600 APEX test series in RAI response 440.339 (Reference 21.5-1.5). The conclusions of this sensitivity study were as follows:

> *The conclusions of this study is that using more nodes in the CMTs represents a way to approximately simulate the CMT thermal stratification effects, to help account for the lack of a CMT thermal stratification model in NOTRUMP. This technique can be used to improve the CMT outlet temperature behavior in small break transients. This CMT noding study supports the conclusions of the independent assessments that are being conducted for the preparation of the summary section for Revision 2 of the NOTRUMP Final Validation Report for AP600. The summary section will indicate that the lack of a CMT thermal stratification model and the coarse noding used lead to significant differences in the CMT outlet temperature and resulting small break transient, but that the continued use of the [4-node]^{a,c} CMT model is acceptable because its effect on the transient is conservative (high core void fraction, delayed ADS).*

The CMT noding utilized for the studies presented herein are shown in Figure 21.5-2.3. One should note that for the transient results presented herein, the use of the increased CMT noding will not have a significant effect due to the time frame over which the CMTs are emptied for the DVI line break simulations. This was subsequently confirmed via the performance of a CMT

noding sensitivity study for the APEX-1000 simulations where the original Reference 21.5-1.4 nodalization was utilized. As expected, revising the CMT noding had little effect on the DEDVI transient simulation results.

The APEX model simulations differ from that utilized in the AP1000 plant design as well. The same differences described above also apply to the modeling differences between the plant model and the OSU model. However, as described above, the CMT noding differences result in a conservative prediction of ADS actuation times and core average void fraction predictions.

Two OSU APEX test simulation results were performed with the Advanced Plant Version of the NOTRUMP computer code. These were:

- Test DBA-02, Double-Ended Direct Vessel Injection Line Break with an ADS-4 single failure on the Pressurizer side (ADS 4-2) side.
- Test DBA-03, Double-Ended Direct Vessel Injection Line Break with an ADS-4 single failure on the non-Pressurizer side (ADS 4-1) side.

The DEDVI line break represents the most severe accident for the AP1000 plant design in that it eliminates a full train of makeup capability. The modeling methodology utilized for the APEX simulations is the same as that utilized for the plant simulations with the following exceptions.

- No passive residual heat exchanger heat transfer [ ]a,c was applied.
- The ADS-4 flow paths were modeled with the [ ]a,c during the transition to non-critical conditions and subsequently the orifice equation for post-critical flow.

The methodology utilized to model the ADS-4 flow paths in the OSU simulations differed from that utilized for the AP1000 plant analysis. In the AP1000 plant simulations, the ADS-4 flow paths are altered from [ ]a,c flow paths to [ ]a,c flow links once non-critical conditions have been reached in both ADS-4 paths. At that time, the FLOAD4 resistance adjustment factor (Reference 21.5-1.5) of [ ]a,c is placed on both ADS-4 flow paths and the transient simulation is continued. Note that the plant and OSU test facility differ in the ADS-4 flow path in that for the plant, the ADS-4 squib valve is the last component in the path and discharges directly to containment. For APEX, the squib valve is represented by a flow venturi with subsequent piping to the ADS-4 separator. This level of detail is not represented in the NOTRUMP model for the APEX test facility. For the APEX simulations performed herein, the ADS-4 flow links in the NOTRUMP model utilized the [ ]a,c. This model was selected based on the results of comparisons of predicted with measured ADS-4 flow. In order to assess the effect of the change in modeling methodology on the AP1000 plant results, the DEDVI line break, assuming atmospheric containment conditions, was re-performed with the same modeling assumption as utilized for the APEX test facility]a,c. The results obtained indicate only a minor change in the predicted ADS-4 behavior and subsequently IRWST injection behavior. To further supplement this conclusion, the APEX-600 test series Double-Ended DVI line break (Test

SB12) was also re-performed utilizing the revised ADS-4 methodology. Again, only minor differences in ADS-4 flow and subsequently IRWST injection times were observed.

The NOTRUMP simulation result comparisons of tests DBA-02 and DBA-03 are presented below.

Comparison of NOTRUMP Simulation to Test Data For Test DBA-02

Figure 21.5-2.4 and Figure 21.5-2.5 compare the pressure at the top of the pressurizer and downcomer regions for the test and the NOTRUMP simulation. The pressure decreases initially due to the blowdown through the break. The depressurization rate slows (and stops for NOTRUMP) when the primary system becomes saturated. Following actuation of ADS-1 at [ ][a,b] seconds in the test (81.3 seconds for NOTRUMP), the depressurization rate increases significantly. The downcomer pressure is provided since this pressure ultimately controls the onset of intact IRWST (IRWST-2) injection. The trends observed in the downcomer pressure closely follow that observed in the pressurizer. The agreement between the test data and the prediction are reasonable since the trends observed are similar.

Figure 21.5-2.6 shows the collapsed liquid level in the pressurizer for the test and the NOTRUMP simulation. The break flow causes a rapid decrease in pressurizer level and empties the pressurizer at approximately [ ][a,b] seconds for the test and 70 seconds for the NOTRUMP simulation. The pressurizer level increases following ADS actuation for both the test and the simulation with NOTRUMP initially refilling faster than the test until ADS-2 actuation. The NOTRUMP simulation collapsed mixture level recovers to slightly lower level following ADS actuation compared to that observed in the test facility. Following ADS-4 actuation, both the test and NOTRUMP simulations indicate a period of continued pressurizer refill until the ADS-4 flow paths become dominant. The collapsed level predicted by NOTRUMP decreases in a similar manner to that observed in the test following ADS-4 actuation. Therefore, the NOTRUMP results are considered to be in reasonable agreement with the test data.

Figure 21.5-2.7 and Figure 21.5-2.8 shows the collapsed liquid levels in CMT-1 and CMT-2 for the test and the NOTRUMP simulation respectively. In the test, CMT-1, which is attached to the broken DVI line begins draining out the break at [ ][a,b] seconds in the test compared to 20 seconds for the NOTRUMP simulation. This can also be seen in the CMT injection flow plots (Figure 21.5-2.9 and Figure 21.5-2.10. As such, the NOTRUMP simulation transitions from re-circulation to draindown mode earlier than observed in the test and subsequently predicts higher injection flows. CMT-2 transitions from recirculation to draindown mode at about [ ][a,b] seconds (150 seconds for NOTRUMP). The comparisons indicate that the NOTRUMP intact CMT drains slightly earlier than observed in the test. This is due to the earlier predicted

emptying of the intact Accumulator (Figure 21.5-2.16 and Figure 21.5-2.28). The conclusions that can be reached are that the NOTRUMP CMT predictions are considered to be in reasonable agreement with the test data. The under-prediction of the transition to CMT-1 draindown mode negligibly impacts the predicted ADS-1 actuation time compared to the test results and is considered reasonable.

Figure 21.5-2.11 through Figure 21.5-2.14 present the collapsed steam generator level comparisons between the test and NOTRUMP simulations. As can be seen, the NOTRUMP results are in good agreement with the test data.

Figure 21.5-2.15 and Figure 21.5-2.16 presents the collapsed liquid levels in accumulator 1 and accumulator 2 for the test and the NOTRUMP simulation. The comparison between the test and the NOTRUMP simulation is considered good for both accumulators with the interruption of accumulator 1 discharge appropriately presented by the NOTRUMP simulation following the transition of CMT-1 from recirculation to draindown mode.

The next series of plots relate to the collapsed and two-phase levels at different locations in the vessel. The trends of the simulation plots are in reasonable agreement with the test up to approximately ADS-2 actuation. Following ADS-2 actuation, the test and NOTRUMP simulations diverge as a result of the test-observed, two-dimensional downcomer behavior which cannot be modeled with the NOTRUMP [ ]^{a,c} downcomer (See Reference 21.5-1.5 for additional details). A review of the core inlet temperature (Figure 21.5-2.29) indicates that the NOTRUMP simulation is predicting sub-cooled conditions whereas the test indicates saturated core entry exist. This can be partly attributed to the lack of two-dimensional downcomer modeling and partly due to heating of the intact DVI injection flow as it impinges on the core barrel. The NOTRUMP model has appropriate heat transfer models from fluid to metal structures in the downcomer fluid node but does not account for the heating of the injected fluid as it impinges onto the core barrel. As such, the injected fluid will retain higher sub-cooling than would be observed in the test facility. To assess the impact of downcomer sub-cooling on the transient simulations, a sensitivity was performed with NOTRUMP in which the intact DVI fluid streams (CMT and Accumulator) were heated to [ ]^{a,c}. The results indicate the divergence observed between the test and NOTRUMP simulation was significantly reduced although not totally eliminated (Figure 21.5-2.17 and Figure 21.5-2.18). This indicates that core inlet sub-cooling, or lack thereof, is partly responsible for the divergence between the NOTRUMP simulation and the test response.

The core collapsed level (Figure 21.5-2.19) plot is in reasonable agreement with the test up to approximately ADS-2 actuation. However, they diverge between [ ]^{a,b} and [ ]^{a,c} seconds due to lack of two-dimensional downcomer modeling and heating of DVI injection flow as

discussed above. The core behavior between the test observed and NOTRUMP predictions re-converge at approximately [ ]^{a,c} seconds. In both cases, the core level initially decreases as inventory is lost from the system. The levels increase following accumulator injection. Once the accumulators empty, the levels continue to increase as a result of CMT-2 injection. Following CMT-2 empty, an injection gap period is encountered. During this period, the core collapsed level slowly decreases until IRWST-2 injection occurs. Since the NOTRUMP simulation predicts a slightly early IRWST-2 injection compared to the test results, it exhibits an earlier recovery than observed in the test. However, the level response is similar between the simulation and test data. A comparison of the core average void fraction is provided as Figure 21.5-2.20. This figure shows lower predicted void fractions during the same divergence period as described above; however, once the conditions re-converge at near [ ]^{a,c} seconds, the NOTRUMP simulation and test data are in reasonable agreement for the remainder of the transient.

The collapsed upper plenum level (Figure 21.5-2.21) indicates that both NOTRUMP and the test simulation have a significant amount of fluid in this region. The upper plenum collapsed level response in the NOTRUMP simulation indicates more sensitivity to the injection gap period than observed in the test (i.e. NOTRUMP predicting a higher inventory loss compared to the test over the injection gap period). The upper plenum two-phase level (Figure 21.5-2.22) follows the same trends as observed in the core and downcomer, that being that the trends are followed reasonably well until ADS-2 through about [ ]^{a,c} seconds. The two-phase level information indicates that both the test and NOTRUMP simulations behave similarly during the injection gap period with both the test and simulation indicating a decrease in mixture level until IRWST-2 injection commences. The NOTRUMP simulation and test data are considered to be in reasonable agreement.

Figure 21.5-2.23 shows the collapsed liquid level in the downcomer for the test and the NOTRUMP simulation. Again there is reasonable agreement between the test and the simulation up to ADS-2 actuation. Following ADS-2 actuation, the test-observed and NOTRUMP-predicted behavior, while similar in trend, diverge. This is once again attributed to the lack of two-dimensional capability in the NOTRUMP [ ]^{a,c} downcomer (see Reference 21.5-1.5 for additional details) and downcomer sub-cooling as described previously. As such, the downcomer levels are predicted reasonably well by NOTRUMP up to ADS-2 actuation and with the noted discrepancy between ADS-2 and [ ]^{a,c} seconds. The comparisons are considered to be reasonable beyond [ ]^{a,c} seconds.

These comparisons demonstrate that the highly ranked PIRT items related to the levels in the core, upper plenum, and downcomer are predicted reasonably well by NOTRUMP up to ADS-2 actuation and with the noted discrepancy between ADS-2 and [ ]^{a,c} seconds. The comparisons are once again considered reasonable beyond [ ]^{a,c} seconds. The discrepancy

period is not considered to be a serious deficiency as the vessel inventory at the critical time of intact IRWST injection is reasonably predicted by NOTRUMP and is consistent with past observations for the DVI line break (Reference 21.5-1.5).

Figure 21.5-2.24 presents a comparison of the vessel mixture inventory between the test and NOTRUMP simulation. As can be seen, the NOTRUMP simulation generally under-predicts the test data with the exception of the period of divergence between ADS-2 and [ ]^{a,c} seconds. This indicates that during the time region of importance, (i.e. Post ADS-4 to IRWST injection) that the NOTRUMP code conservatively predicts the vessel conditions. The slightly early IRWST-2 injection, predicted by NOTRUMP, is clearly seen in this figure as the point at which the minimum inventory is predicted. This indicates that the NOTRUMP code is performing reasonably.

Figure 21.5-2.25 shows the integrated mass flow through ADS stage-4 for the test and the NOTRUMP simulation. These curves show that the ADS stage-4 flow is slightly over-predicted by NOTRUMP after about 450 seconds. The flows match reasonably well as indicated by the parallel behavior of the integrated flow curves and the observed trends. This agreement in the slope of the curves demonstrates that the PIRT highly ranked items related to ADS stage-4 (critical flow, two-phase pressure drop, and valve loss coefficients) are predicted reasonably by NOTRUMP.

Figure 21.5-2.26 shows the integrated mass flow out of the break for the test and the NOTRUMP simulation. For this simulation, the NOTRUMP model applied a discharge coefficient of [ ]^{a,b} to more accurately represent the results observed in the test. Differences in modeling of the break and break measurement system in the test and NOTRUMP simulations also affects the results. This is described in more detail in the response to RAI.440.721(d) (Reference 21.5-1.8). Although the integrated break flow is slightly over-predicted by NOTRUMP, the general trends of the test break flow are similar to the prediction. This demonstrates that the PIRT highly ranked item of break critical flow can be predicted by NOTRUMP.

Figure 21.5-2.27 and Figure 21.5-2.28 show the total DVI line flow rates between the NOTRUMP simulation and the test for DVI line 1 and DVI line 2 respectively. The simulation data, provided for DVI line 1, represents the break flow from the DVI side piping of the DEDVI break. As can be seen, although the trends are predicted, the behavior of the ruptured DVI line (DVI-1) over-predicts the initial CMT draindown rate as described earlier. This causes an early prediction of ADS actuation for the NOTRUMP simulation compared to the test prediction. However, this is assessed to have a minimal impact on the results. As such, the results are considered reasonable. Figure 21.5-2.28 presents the intact side DVI line flow (DVI-2) for both

the test and NOTRUMP simulation. The results indicate that the intact side DVI flow is predicted well by NOTRUMP. Since this path represents the makeup source, it represents an important characteristic that is well predicted by the NOTRUMP simulation.

Figure 21.5-2.29 and Figure 21.5-2.30 present the core inlet and core outlet temperatures between the test and NOTRUMP simulation respectively. The core inlet temperature is approximately the same as the simulation until approximately 150 seconds of the transient, while the outlet temperature is predicted well. After 300 seconds, the core inlet fluid temperature is over-predicted and is likely due to the removal of the PRHR model from the NOTRUMP simulation to conservatively account for the potential accumulation of non-condensable gases in the PRHR tubes, which cannot be directly modeled with NOTRUMP. As such, the NOTRUMP comparisons are considered reasonable.

Comparison of NOTRUMP Simulation to Test Data For Test DBA-03

Figure 21.5-2.31 and Figure 21.5-2.32 compare the pressure at the top of the pressurizer and downcomer regions for the test and the NOTRUMP simulation. The pressure decreases initially due to the blowdown through the break. The depressurization rate slows (and stops for NOTRUMP) when the primary system becomes saturated. Following actuation of ADS-1 at [ ][a,b] seconds in the test (84.4 seconds for NOTRUMP), the depressurization rate increases significantly. NOTRUMP predicts a higher pressure than observed in the test for most of the time. The trends observed in the pressurizer are also observed in the downcomer pressure response as well. As such, the agreement between the test data and the prediction is considered to be reasonable for the primary pressure response with the trends of the data being similar to that observed in the test.

Figure 21.5-2.33 shows the collapsed liquid level in the pressurizer for the test and the NOTRUMP simulation. The break flow causes a rapid decrease in pressurizer level and empties the pressurizer at approximately [ ][a,b] seconds for the test and 70 seconds for the NOTRUMP simulation. The pressurizer level increases following ADS actuation for both the test and the simulation with NOTRUMP initially refilling faster than the test until ADS-2 actuation. The NOTRUMP simulation collapsed mixture level recovers to approximately the same level following ADS actuation. Following ADS-4 actuation, both the test and NOTRUMP simulations indicate a period of continued pressurizer refill until the ADS-4 flow paths become the dominant depressurization paths. The pressurizer collapsed level decreases in a similar manner following ADS-4 actuation for both the simulation and the test data. Therefore, the NOTRUMP results are considered to be in reasonable agreement with the test data.

Figure 21.5-2.34 and Figure 21.5-2.35 shows the collapsed liquid levels in CMT-1 and CMT-2 for the test and the NOTRUMP simulation respectively. In the test, CMT-1, which is attached to the broken DVI line begins draining out the break at [ ][a,b] seconds in the test compared to 20 seconds for the NOTRUMP simulation. This can also be seen in the CMT injection flow plots (Figure 21.5-2.36 and Figure 21.5-2.37). As such, the NOTRUMP simulation transitions from re-circulation to draindown mode earlier than observed in the test and subsequently predicts higher injection flows. The comparisons indicate that the NOTRUMP intact CMT drains earlier than observed in the test. This is due to both the earlier predicted emptying of the intact Accumulator by the NOTRUMP simulation (Figure 21.5-2.43 and Figure 21.5-2.53) and the earlier IRWST injection observed in the test. The conclusions that can be reached are that the NOTRUMP results for the CMT behavior is considered to be in reasonable agreement with the test data. The under-prediction of the transition to CMT-1 draindown mode negligibly impacts the predicted ADS-1 actuation time compared to the test results and is considered reasonable.

Figure 21.5-2.38 through Figure 21.5-2.41 present the collapsed steam generator level comparisons between the test and NOTRUMP simulations. As can be seen, the NOTRUMP results are in good agreement with the test data.

Figure 21.5-2.42 and Figure 21.5-2.43 shows the collapsed liquid levels in accumulator 1 and accumulator 2 for the test and the NOTRUMP simulation. As can be seen, the intact accumulator injection characteristics differ significantly between the NOTRUMP simulation and the test. When one reviews the injection characteristics compared to test DBA-02, the intact accumulator differs in an unexpected fashion. Since the differences between test DBA-02 and test DBA-03 are limited to the ADS-4 failure location, the changes expected, between test DBA-02 and DBA-03, should occur following ADS-4 actuation. However, as can be seen the transients diverge prior to this time. The comparison between the test and the NOTRUMP simulation is considered good for accumulator 1; however, the comparison for accumulator 2 is considered minimal. 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. This is particularly evident in the time period prior to IRWST injection during which the NOTRUMP vessel mass is conservatively predicted relative to the test (Figure 21.5-2.49).

The next series of plots relate to the collapsed and two-phase levels at different locations in the vessel. The trends of the simulation plots are in reasonable agreement with the test up to approximately ADS-2 actuation. Following ADS-2 actuation, the test and NOTRUMP simulations diverge as a result of the test-observed, two-dimensional downcomer behavior which cannot be modeled with the NOTRUMP [ ][a,c] downcomer (See Reference 21.5-1.5

for additional details) and the downcomer sub-cooling as described in the previous discussion of test DBA-02.

The core collapsed level (Figure 21.5-2.44) plot is in reasonable agreement with the test up to approximately ADS-2 actuation. However, they diverge time between [ ]^{a,b} and [ ]^{a,c} seconds as discussed above. The core behavior between the test observed and NOTRUMP predictions re-converge at approximately [ ]^{a,c}. In both cases, the core level initially decreases as inventory is lost from the system. The levels increase following accumulator injection. Once the accumulators empty, the levels continue to increase as a result of CMT-2 injection. For this case, the test indicates that continuous injection will occur while the NOTRUMP simulation indicates an injection gap period will occur. During this predicted injection gap, the NOTRUMP core mixture level decreases slightly until IRWST-2 injection occurs at which time a core level recovery occurs. A comparison of the core average void fraction is provided as Figure 21.5-2.45. This figure shows lower predicted void fractions during the same divergence period as described above; however, once the conditions re-converge at near [ ]^{a,c} seconds, the NOTRUMP simulation and test data are in reasonable agreement for the remainder of the transient.

The collapsed upper plenum level (Figure 21.5-2.46) indicates that both NOTRUMP and the test simulation have a significant amount of fluid in this region as was observed in test DBA-02. The upper plenum collapsed level response in the NOTRUMP simulation indicates the same behavior as observed in the test. The upper plenum two-phase level (Figure 21.5-2.47) follows the same trends as observed in the core and downcomer, that being that the trends are followed reasonably well until ADS-2 through about [ ]^{a,c} seconds. The comparisons also indicate that that the NOTRUMP simulation does not recover the two-phase level as quickly as a result of the predicted injection gap as compared to the test observed conditions. . Following IRWST injection, the test indicates a more rapid increase in the two-phase mixture level as compared to the NOTRUMP simulation, which increases more slowly. As such, the two-phase mixture level is conservatively predicted by NOTRUMP following intact IRWST injection and is considered reasonable.

Figure 21.5-2.48 shows the collapsed liquid level in the downcomer for the test and the NOTRUMP simulation. Again there is reasonable agreement between the test and the simulation up to ADS-2 actuation. Following ADS-2 actuation, the test-observed and NOTRUMP-predicted behavior diverges. This is attributed to the lack of two-dimensional capability in the NOTRUMP [ ]^{a,c} downcomer (see Reference 21.5-1.5 for additional details) and downcomer sub-cooling as described previously. As such, the downcomer levels are predicted reasonably well by NOTRUMP up to ADS-2 actuation and with the noted

discrepancy between ADS-2 and [ ] ^{a,c} seconds. The comparisons are considered to be reasonable beyond [ ] ^{a,c} seconds.

These comparisons demonstrate that the highly ranked PIRT items related to the levels in the core, upper plenum, and downcomer are predicted reasonably well by NOTRUMP up to ADS-2 actuation and with the noted discrepancy between ADS-2 and [ ] ^{a,c} seconds. The comparisons are once again considered reasonable beyond [ ] ^{a,c} seconds.

Figure 21.5-2.49 presents a comparison of the vessel mixture inventory between the test and NOTRUMP simulation. As can be seen, the NOTRUMP simulation generally under-predicts the test data with the exception of the period of divergence between ADS-2 and [ ] ^{a,c} seconds. This indicates that during the time region of importance, (i.e. Post ADS-4 to IRWST injection) that the NOTRUMP code conservatively predicts the vessel conditions. The delay in the predicted IRWST-2 injection is clearly seen in this figure as the point at which the minimum inventory is predicted. This indicates that the NOTRUMP code is performing reasonably.

Figure 21.5-2.50 shows the integrated mass flow through ADS stage-4 for the test and the NOTRUMP simulation. These curves show that the ADS stage-4 flow is slightly over-predicted by NOTRUMP. This comparison demonstrates that the PIRT highly ranked items related to ADS stage-4 (critical flow, two-phase pressure drop, and valve loss coefficients) are predicted reasonably by NOTRUMP.

Figure 21.5-2.51 shows the integrated mass flow out of the break for the test and the NOTRUMP simulation. For this simulation, the NOTRUMP model applied a discharge coefficient of [ ] ^{a,b} to more accurately represent the results observed in the test. Differences in modeling of the break and break measurement system in the test and NOTRUMP simulations also affects the results. This is described in more detail in the response to RAI 440.721(d) (Reference 21.5-1.8). The test observed conditions indicate additional liquid discharge occurring at approximately [ ] ^{a,c} seconds as a result of the higher observed downcomer mixture level compared to the NOTRUMP predicted results. In addition, since the test indicates earlier IRWST injection, compared to the NOTRUMP simulation, the break flows follow this trend as well. The general trends of the test break flow are similar to the NOTRUMP prediction. This demonstrates that the PIRT highly ranked item of break critical flow can be reasonably predicted by NOTRUMP.

Figure 21.5-2.52 and Figure 21.5-2.53 show the total DVI line flow rates between the NOTRUMP simulation and the test for DVI line 1 and DVI line 2 respectively. The simulation data, provided for DVI line 1, represents the break flow from the DVI side piping of the DEDVI break. As can be seen, although the trends are predicted, the behavior of the ruptured DVI line

(DVI-1) over-predicts the initial CMT draindown rate. This results in an early prediction of ADS actuation for the NOTRUMP simulation compared to the test prediction. However, this is assessed to have a minimal impact on the results. As such, the results are considered reasonable. Figure 21.5-2.53 presents the intact side DVI line flow (DVI-2) for both the test and NOTRUMP simulation. The results indicate that the intact side DVI flow is predicted reasonably by NOTRUMP.

Figure 21.5-2.54 and Figure 21.5-2.55 present the core inlet and core outlet temperatures between the test and NOTRUMP simulation respectively. The core inlet temperature is approximately the same as the simulation until approximately 150 seconds of the transient, while the outlet temperature is predicted well. After 300 seconds, the core inlet fluid temperature is over-predicted and is likely due to the removal of the PRHR model from the NOTRUMP simulation to conservatively account for the potential accumulation of non-condensable gases in the PRHR tubes, which cannot be directly modeled with NOTRUMP. As such, the NOTRUMP comparisons to considered reasonable for the modeling capability available.

Overall Conclusions

The following conclusions can be reached by reviewing the NOTRUMP predicted response compared to the test observed conditions:

- NOTRUMP predicts the effect of the ADS-4 single failure location as observed in the test.
- NOTRUMP conservatively predicts vessel inventory during the ADS-4 to IRWST injection period.
- NOTRUMP predicts the pressurizer mixture level performance reasonably well.
- NOTRUMP predicts IRWST injection flow reasonably well.
- The divergence of vessel inventory between ADS-2 actuation to approximately [ ]^{a,c} seconds is a multi-dimensional effect and sub-cooling effect which can not be properly modeled by NOTRUMP; however, the duration of this period is small and the ADS-4 to IRWST injection period is considered to be reasonable.
- The results indicate that the NOTRUMP code performs reasonably compared to tests designed specifically for comparisons to the AP1000 plant design. As such, it continues to be applicable for analyses of SBLOCA events for the AP1000 plant design.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

The information from this RAI response will be incorporated into WCAP-15644, AP1000 Code Applicability Report.

References:

21.5-1.1 DSER Open Item 21.5-1 Response.

21.5-1.2 OSU-APEX-03002, Revision 0, OSU Facility Description Report for AP1000 Simulation Series, K. C. Abel, et al., Oregon State University, Department of Nuclear Engineering, May 12, 2003.

21.5-1.3 OSU-APEX-03001, Revision 0, Scaling Assessment for the Design of the OSU APEX-1000 Test Facility, J. Reyes, et al., Oregon State University, Department of Nuclear Engineering, May 12, 2003.

21.5-1.4 WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 2, Section 8.0, August 1998.

21.5-1.5 WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 3, Appendix-A, RAI-440.796F, Part a, August 1998.

21.5-1.6 WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 1, Section 6, August 1998.

21.5-1.7 WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 3, RAI 440-339, August 1998.

21.5-1.8 WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 3, RAI 440-721(d), August 1998.

21.5-1.9 WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 3, RAI 440-721(f), August 1998.