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Your ref Docket No. 52-006
Our ref. DCP/NRC1559

March 31, 2003

SUBJECT: Transmittal of Westinghouse Responses to US NRC Requests for Additional Information on the AP1000 Application for Design Certification

This letter transmits the Westinghouse responses to NRC Requests for Additional Information (RAI) regarding our application for Design Certification of the AP1000 Standard Plant. A list of the RAI responses that are transmitted with this letter is provided in Attachment 1. Attachment 2 provides the RAI responses.

Please contact me if you have questions regarding this submittal.

Very truly yours,

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

/Attachments

1. Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1559"
2. Westinghouse Non-Proprietary Response to US Nuclear Regulatory Commission Requests for Additional Information dated March 2003

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March 31, 2003

Attachment 1

"List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1559"

Attachment 1

Table 1

"List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1559"

210.055, Rev. 1	630.028, Rev. 1
210.028, Rev. 1	630.052, Rev. 1
210.001, Rev. 1	720.013, Rev. 1
251.012, Rev. 1	720.021, Rev. 1
251.021, Rev. 1	720.025, Rev. 1
410.019, Rev. 1	720.039, Rev. 1
440.036, Rev. 1	720.043, Rev. 1
440.154, Rev. 2	720.048, Rev. 1
440.157, Rev. 1	720.050, Rev. 1
440.160, Rev. 1	720.060, Rev. 1
440.171, Rev. 1	720.092, Rev. 1
440.186, Rev. 0	

Attachment 2

Westinghouse Non-Proprietary Response to US Nuclear Regulatory Commission
Requests for Additional Information dated March 2003

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.001 (Response Revision 1)

Revision 0 Question:

Reference, Volume 6, Section 3.9.2.3, Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions, Pg. 3.9-31, last paragraph:

Westinghouse (W) proposes that the assessment of RPV internals flow-induced vibrational response is done using a combination of analysis and testing, as specified in Regulatory Guide (R.G.) 1.20. However, W also proposes that the entire vibration assessment program, including the predictive analysis portion, will be performed by the Combined License (COL) applicant. This proposal is repeated in DCD Section 3.9.8.1 (Volume 6, Pg. 3.9-93) citing consistency with R.G. 1.20 as a basis for deferral of the performance of the entire vibration assessment program to the COL applicant.

The NRC staff is not in complete agreement with this proposal for the following reasons. Title 10 of the Code of Federal Regulations (10 CFR) Part 52.47(a)(2) requires that applications for standard design certification must contain a level of design information sufficient to enable the Commission to reach a final conclusion on all safety questions associated with the design before the certification is granted. Delaying the predictive analysis portion of the vibration assessment program to the COL applicant stage of plant construction does not provide the staff with a level of design information sufficient to reach a final conclusion regarding adequacy of the RPV internals design. Conformance with R.G. 1.20 alone, does not necessarily fulfill the requirements of 10 CFR 52.47(a)(2) for certification of the adequacy of the standard design of the RPV internals, primarily because the R.G. 1.20 scheduling requirements for the submittal of analytical results to the staff occurs much too late to support the standard design certification process.

The staff's position on this issue is that the detailed, predictive analysis portion of the RPV internals flow-induced vibration analysis program should be provided for staff review during the design certification process, and not be deferred to the COL applicant stage of actual plant construction. It is recognized that the other phases of the comprehensive R.G. 1.20 vibration assessment program, i.e., vibration measurement and physical inspection, must be done later by the COL applicant to confirm the predictive analysis results. However, the staff considers the results of the predictive analysis phase of this program to be the kind of detailed information necessary for the staff to make a determination of adequacy of the AP1000 RPV internals design for purposes of final design certification.

Please provide technical documentation of the predictive analysis phase of the vibration assessment program. The technical details should be provided with descriptions of the analytical methods used including computer models, results of the analyses summarized in tabular format, and comparisons of calculated stresses to ASME Code allowables for the major components of the RPV internals design. An example of the presentation of the type of analytical data requested is W topical report WCAP-14761, "AP600 Reactor Internals Flow-Induced Vibration Assessment Program." This type of topical report would also be appropriate for presentation of key details of the AP1000 prototype RPV internals design necessary for staff review at the standard design certification stage.

Revision 0 Westinghouse Response:

A flow-induced vibration assessment of the AP1000 reactor vessel internals will be performed and the results will be documented in an AP1000 Vibration Assessment Report which will be provided to the NRC to support AP1000 Design Certification. This report will be similar in scope to WCAP-14761, "AP600 Reactor Internals Flow-Induced Vibration Assessment Program". The predictive analysis portion of the reactor vessel internals flow-induced vibration analysis program will be included in the report.

The AP1000 Vibration Assessment Report (WCAP-15949) will be referenced in DCD Section 3.9.2.3 and will be included in the Referenced Material list in DCD Table 1.6-1.

The reactor internals vibration assessment will be deleted from the Combined License Information in DCD Section 3.9.8.1 and from the Combined License Information listed in DCD Table 1.8-2.

Design Control Document (DCD) Revision:

From DCD page 3.9-31: (These changes have been incorporated into DCD Revision 3)

3.9.2.3 Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow-induced excitation are complex and not readily ascertained by analytical means alone. Assessment of vibrational response is done using a combination of analysis and testing. Comparisons of results obtained from reference plant vibration measurement programs have been used to confirm the validity of scale model tests and other prediction methods as well to confirm the adequacy of reference plant internals regarding flow induced vibration. The flow-induced vibration assessment **is documented in WCAP-15949 (Reference 18).**~~will be performed by the Combined License applicant.~~

From DCD Page 3.9-92: (These changes have been incorporated into DCD Revision 3)

3.9.8 Combined License Information

3.9.8.1 Reactor Internals Vibration Assessment and Predicted Response

~~The flow induced vibration assessment of the reactor internals will be performed by the Combined License applicant. The assessment will be consistent with the guidelines of Regulatory Guide 1.20.~~ Information including predicted vibration response and allowable response will be provided prior to the preoperational vibration testing of the first AP1000 consistent with the guidance of Regulatory Guide 1.20.

From DCD page 3.9-94, Section 3.9.9 References: (These changes have been incorporated into DCD Revision 3)

18. "AP1000 Reactor Internals Flow Induced Vibration Assessment Program," WCAP-15949-P (Proprietary) and WCAP-15949-NP (Non-proprietary), November 2002.

From DCD page 1.6-5, Table 1.6-1: (These changes have been incorporated into DCD Revision 3)

Table 1.6-1 (Sheet 4 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
3.8	WCAP-13891	AP600 Automatic Depressurization System Phase A Test Data Report, May 1994
	WCAP-14324	Final Data Report for ADS Phase B1 Tests, April 1995
	WCAP-15613	AP1000 PIRT and Scaling Assessment, March 2001
3.9	WCAP-7765-AR	Westinghouse PWR Internals Vibrations Summary Three-Loop Internals Assurance, November 1973
	WCAP-8766 (P) WCAP-8780	Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant, May 1976
	WCAP-8516-P (P) WCAP-8517	UHI Plant Internals Vibrations Measurement Program and Pre- and Post-Hot Functional Examinations, March 1975
	WCAP-10846 (P)	Doel 4 Reactor Internals Flow-Induced Vibration Measurement Program, March 1985
	WCAP-10865 (P) WCAP-10866	South Texas Plant (TGX) Reactor Internals Flow-Induced Vibration Assessment, February 1985
	WCAP-8708-P-A (P) Volumes 1 and 2 WCAP-8709-A Volumes 1 and 2	MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, February 1976
	WCAP-8446 (P) WCAP-8449	17x17 Drive Line Components Tests - Phase 1B 11, 111 D-Loop Drop and Deflection, December 1974
	WCAP-9693 (P)	Investigation of Feedwater Line Cracking in Pressurized Water Reactor Plants, June 1980
	WCAP-15949-P (P) **WCAP-15949-NP**	**AP1000 Reactor Internals Flow Induced Vibration Assessment Program, November 2002**

(P) Denotes Document is Proprietary

From DCD page 1.8-12, Table 1.8-2: (These changes have been incorporated into DCD Revision 3)

Table 1.8-2 (Sheet 2 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
2.5-13	Subsurface Instrumentation	2.5.4.6.11
2.5-14	Stability of Slopes	2.5.5
2.5-15	Embankments and Dams	2.5.6
3.3-1	Wind and Tornado Site Interface Criteria	3.3.3
3.4-1	Site-Specific Flooding Hazards Protective Measures	3.4.3
3.5-1	External Missile Protection Requirements	3.5.4
3.6-1	Pipe Break Hazards Analysis	3.6.4.1
3.6-2	Leak-Before-Break Evaluation of as-Designed Piping	3.6.4.2
3.6.3	Leak-Before-Break Evaluation of as-Built Piping	3.6.4.3
3.7-1	Seismic Analysis of Dams	3.7.5.1
3.7-2	Post-Earthquake Procedures	3.7.5.2
3.7-3	Seismic Interaction Review	3.7.5.3
3.7-4	Reconciliation of Seismic Analyses of Nuclear Island Structures	3.7.5.4
3.7-5	Seismic Analyses of Nuclear Island Structures at Soil Sites	3.7.5.5
3.8-1	Containment Vessel Design Adjacent to Large Penetrations	3.8.6.1
3.8-2	Passive Containment Cooling System Water Storage Tank Examination	3.8.6.2
3.8.3	Design Summary Reports - Structures	3.8.6.3
3.8-4	As-Built Summary Report	3.8.6.4
3.9-1	**Reactor Internal Vibration Response**	**3.9.8.1**
3.9-2	Design Specification and Reports	3.9.8.2
3.9-3	Snubber Operability Testing	3.9.8.3
3.9-4	Valve Inservice Testing	3.9.8.4
3.9-5	Surge Line Thermal Monitoring	3.9.8.5
3.9-6	Piping Benchmark Program	3.9.8.6

Table 1.8-2 (Sheet 2 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
3.10-1	Experience-Based Qualification	3.10.6
3.11-1	Equipment Qualification File	3.11.5
4.2-1	Changes to Reference Reactor Design	4.2.5

PRA Revision:

None

NRC Additional Comments on WCAP-15949:

The following items in WCAP-15949 need further clarification:

1. (Editorial) Pg. 3-5, Figure 3-1: Three different reactor lower internals configurations are shown, but not specifically identified. AP1000 appears to be the one on the right; please identify the other two.

2. (Editorial) Pg. 5-16, Table 5-6. The Calculated Maximum (Stress) Gradient reported for the guide tubes, top location at 409 Hz, is not consistent with the other data reported. Please confirm that this should read 0.54e-4 psi/in, instead of 0.54 e04 psi/in.

3. (Editorial) Pg. 6-4, second line. The units for pump rotating frequency are expressed in mils. Please clarify.

4. Pg. 6-5, first bullet, and pg. 6-8, last paragraph. The acronym 'CFX' is used. Please provide additional explanation of what CFX refers to; e.g., is it a computer code, a computer model of a specific plant configuration, etc.?

5. (Editorial) Pg. 6-7, fifth paragraph. The minimum high cycle fatigue margin (3.16) reported for the guide tubes is not consistent with data listed in Table 6-5. Please clarify.

6. Pg. 6-9, Section 6.4.2.1. The calculation of required bolt preload concludes that the resulting needed preload per bolt is about 28,600 lb. Table 6-8, Net Preload Acting on Lower Flange of Core Shroud, reports a net bolt preload of 33,022 lbs, suggesting that this is the total preload for all sixteen bolts, which does not appear to be consistent with the preload required per bolt, from pg. 6-9. Please clarify.

7. Pg.6-13, Table 6-3. The guide tube (location H-14) steady flow load calculated by the CFD analysis is significantly lower than the corresponding load from the scale model results. Please provide some justification for this result, which is not consistent with other data trends reported in Table 6-3.

8. (Editorial) Pg. 7-2, fifth bullet. The locations of the two strain gages discussed are not included in Fig. 7-1, Location of Transducers for the AP1000 Preoperational Vibration Measurement Program. Please clarify.

9. Pg. 7-11, Fig. 7-3. These figures are graphically confusing; what is intended? what is the frame of reference, or coordinate system?

10. (Editorial) Superscripts, including the lower case letters 'a', 'b', and 'c', are used throughout the text and tables without accompanying footnotes explaining their meaning. Examples include Pg. 1-1, last paragraph, Pgs. 3-4, 4-2, and 6-2, Tables 4-2 through 4-7, Tables 5-3 through 5-6, etc. Please clarify the meaning of these superscripts.

Westinghouse Response to Additional Comments on WCAP-15949:

The changes indicated below will be incorporated in Revision 1 of WCAP-15949.

1. Page 3-5: Identification of the plant design for each of the three reactor internals are, left to right: 3 Loop 12 ft core, 3 Loop 14 ft core, and AP1000.

2. Page 5-16: The correct number is 0.54e-4 psi/in.

3. Page 6-4: The correct unit is Hertz (Hz).

4. Page 6-5 and 6-8: CFX is a computer code from AEA Technology Engineering Software Limited.

5. Page 6-7: The minimum high cycle fatigue margin for the guide tubes is 3.7 as given in Table 6-5.

6. Page 6-15: Table 6-8 title will be changed to indicate the given preload is per bolt. Also the sign on the hydraulic drag force on the core shroud will be corrected.

7. Page 6-13, Table 6-3: In general the ratios of CFD to measured 1/7 scale model mean loads vary between 0.83 and 1.19 or less than ± 20%. Given that the CFD to measured prototype ratios are well above unity, and the margins of the guide tubes using CFD are greater than 3, then the 17% uncertainty does not significantly change the design margin.

   This explanation will be added to the discussion on page 6-7.

8. Page 7-9: Figure 7-1 will be revised to show the location of the strain gages on the outer wall on the core barrel at the connection to the lower core support plate.

9. Page 7-11: The purpose of Figure 7-3 is to show that the core shroud beam modes are very similar both with and without the core included in the model. Figure 7-3 will be revised to show a different angle of the core shroud model to help clarify the mode shape. An indication of the coordinate system will also be added to the figure.

10. Page 1-1: "[ ]" indicates that the enclosed data is proprietary data. The superscripts are defined as follows:
"a" – Information reveals the distinguishing aspects of a process or component, structure, tool, method, etc.
"b" – Information consists of supporting data relative to a process or component, structure, tool, method, etc.
"c" – Information, if used by a competitor, would reduce competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, etc. of a similar product.
"g" – Information is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner

The above information will be added as a footnote to page 1-1.

WCAP-15949 Revision:

The attached marked-up pages reflect the changes to the WCAP described above.

Design Control Document (DCD) Revision:

From DCD Revision 3 page 1.6-5, Table 1.6-1:

Table 1.6-1 (Sheet 4 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
3.8	WCAP-13891	AP600 Automatic Depressurization System Phase A Test Data Report, May 1994
	WCAP-14324	Final Data Report for ADS Phase B1 Tests, April 1995
	WCAP-15613	AP1000 PIRT and Scaling Assessment, March 2001
3.9	WCAP-7765-AR	Westinghouse PWR Internals Vibrations Summary

Table 1.6-1 (Sheet 4 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
		Three-Loop Internals Assurance, November 1973
	WCAP-8766 (P) WCAP-8780	Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant, May 1976
	WCAP-8516-P (P) WCAP-8517	UHI Plant Internals Vibrations Measurement Program and Pre- and Post-Hot Functional Examinations, March 1975
	WCAP-10846 (P)	Doel 4 Reactor Internals Flow-Induced Vibration Measurement Program, March 1985
	WCAP-10865 (P) WCAP-10866	South Texas Plant (TGX) Reactor Internals Flow-Induced Vibration Assessment, February 1985
	WCAP-8708-P-A (P) Volumes 1 and 2 WCAP-8709-A Volumes 1 and 2	MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, February 1976
	WCAP-8446 (P) WCAP-8449	17x17 Drive Line Components Tests - Phase 1B 11, 111 D-Loop Drop and Deflection, December 1974
	WCAP-9693 (P)	Investigation of Feedwater Line Cracking in Pressurized Water Reactor Plants, June 1980
	WCAP-15949-P (P) WCAP-15949-NP	AP1000 Reactor Internals Flow Induced Vibration Assessment Program, ~~November 2002~~**April 2003**

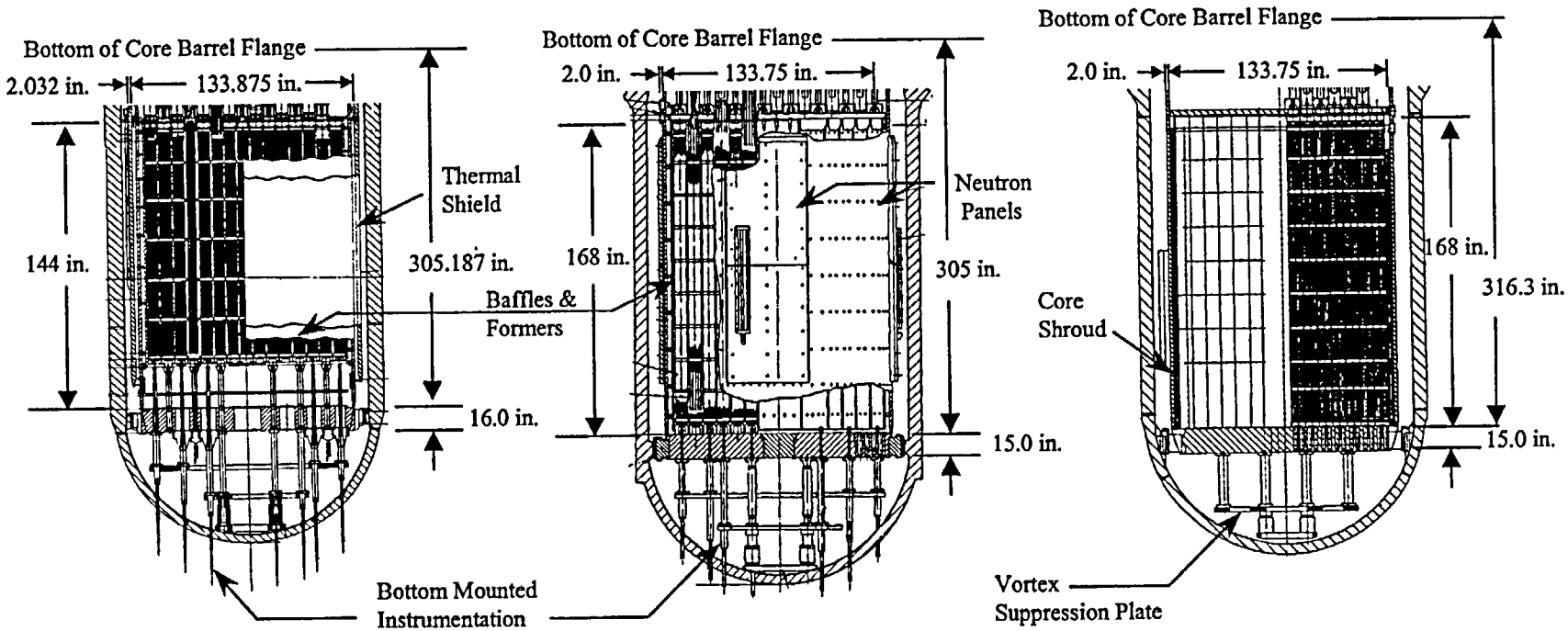
From DCD Revision 3 page 3.9-94, Section 3.9.9 References:

1. "AP1000 Reactor Internals Flow Induced Vibration Assessment Program," WCAP-15949-P, **Revision 1** (Proprietary) and WCAP-15949-NP, **Revision 1** (Non-proprietary), ~~November 2002~~**April 2003**.

PRA Revision:

None

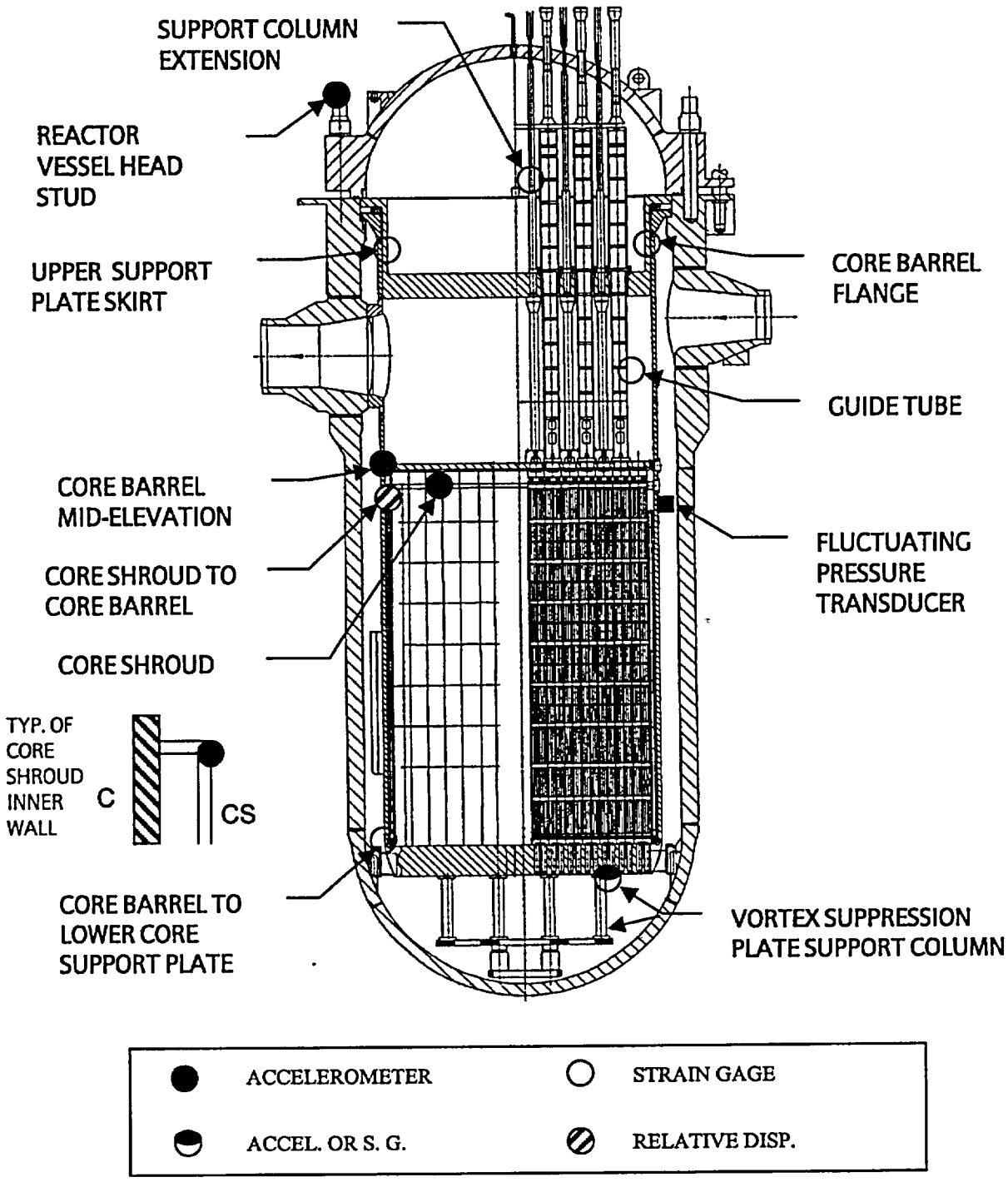
(P) Denotes Document is Proprietary



3 Loop, 12 ft. Core

3 Loop, 14 ft. Core (XL)

AP1000


SUPPORT COLUMN EXTENSION
REACTOR VESSEL HEAD STUD
UPPER SUPPORT PLATE SKIRT
CORE BARREL FLANGE
GUIDE TUBE
CORE BARREL MID-ELEVATION
CORE SHROUD TO CORE BARREL
CORE SHROUD
FLUCTUATING PRESSURE TRANSDUCER
TYP. OF CORE SHROUD INNER WALL
C
CS
CORE BARREL TO LOWER CORE SUPPORT PLATE
VORTEX SUPPRESSION PLATE SUPPORT COLUMN
ACCELEROMETER
STRAIN GAGE
ACCEL. OR S. G.
RELATIVE DISP.

REPLACE WITH ATTACHED FIGURE

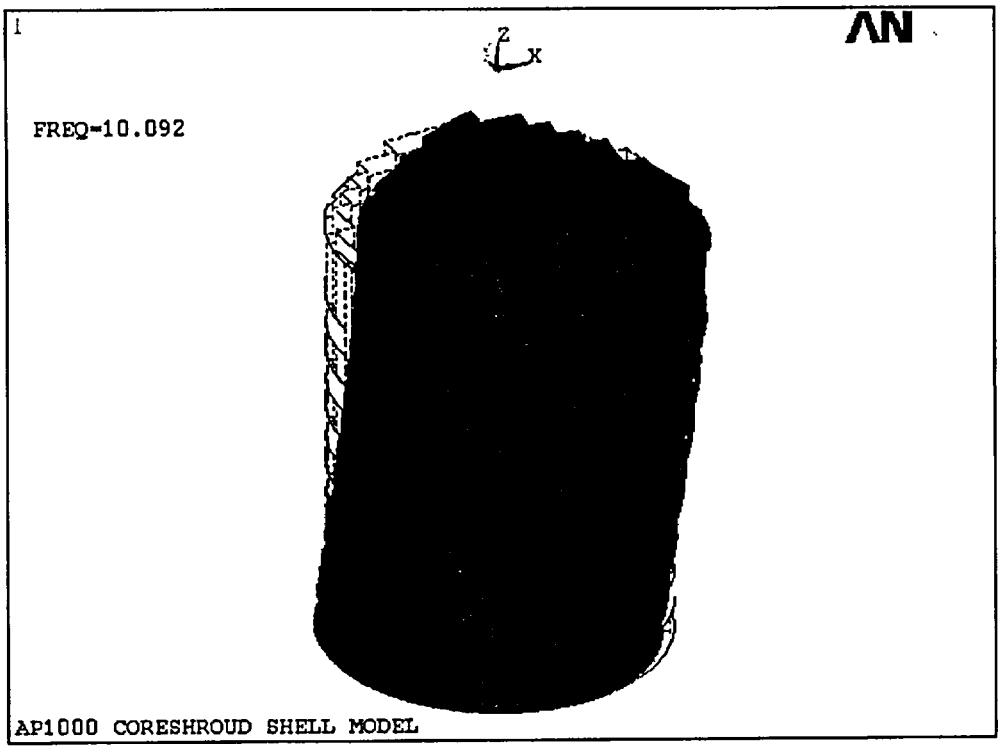
FREQ =10.092
(with core)

[redacted]

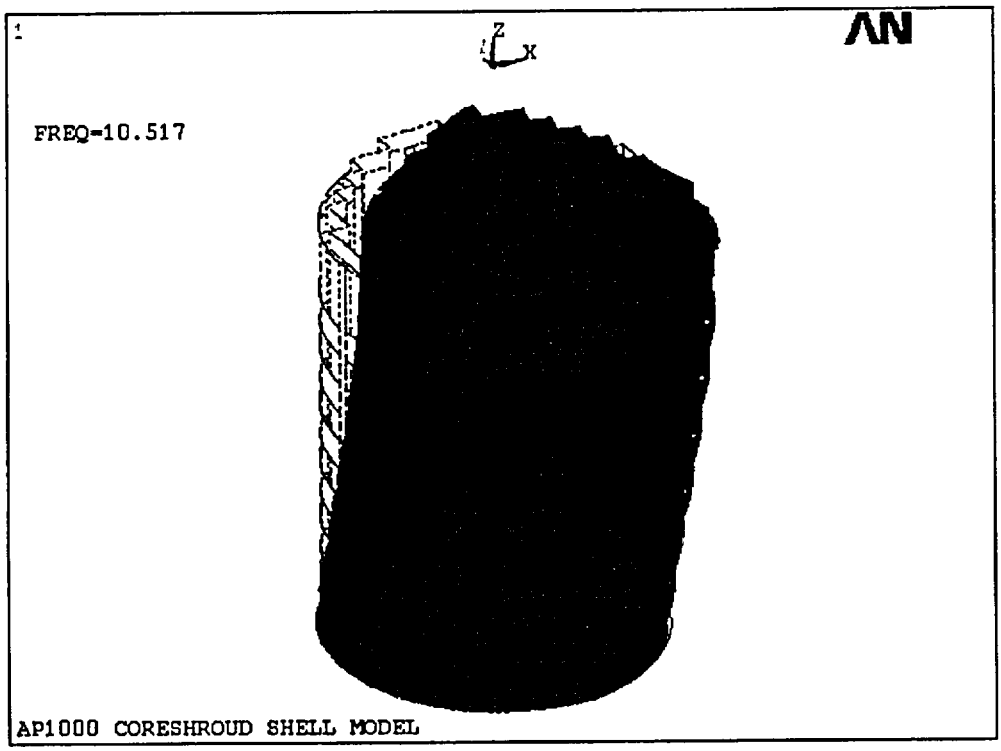
FREQ =10.517
(without core)

[redacted]

Figure 7-3 Core Shroud Beam Modes With and Without Core



With Core



Without Core

1 INTRODUCTION

With respect to the reactor internals preoperational test program, the first AP1000 plant reactor vessel internals are classified as prototype as defined in the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.20, Revision 2. AP1000 reactor vessel internals do not represent a first-of-a-kind or unique design based on the general arrangement, design, size, or operating conditions. The units referenced as supporting the AP1000 reactor vessel internals design features and configuration have successfully completed vibration assessment programs, including vibration measurement programs. These units have subsequently demonstrated extended satisfactory inservice operation. Additional background on the Westinghouse position with regard to NRC Regulatory Guide 1.20 is provided in Westinghouse Nuclear Safety Position Papers (References 1-2 and 1-3).

The vibration assessment approach for the AP1000 is believed to meet the intent of NRC Regulatory Guide 1.20 and is similar to the approach taken by Westinghouse on previous plants. Westinghouse performed a similar vibration assessment in support of the AP600 Design Certification. The AP600 vibration assessment was documented in Reference 1-4. This report provides a similar format and content as was provided for the AP600 assessment.

The purpose of this AP1000 reactor internals vibration assessment program is to demonstrate structural adequacy with respect to flow- and pump- induced vibrations. Estimates of flow-induced vibration levels and forces (or relative values) of the AP1000 plant are made on the basis of scale model tests, tests on prototype reactors, and results of analytical calculations. Based on this information, the vibratory behavior of the reactor internals is well characterized and the vibration amplitudes are sufficiently low for structural adequacy of the components.

The H. B. Robinson no. 2 plant has historically been established as the prototype design for three-loop plant internals and was instrumented and tested during hot functional testing. The test and analysis results of the three-loop configuration of Reference 1-3 demonstrate that the vibration levels of the reactor internals components are low and that the vibrations are adequately characterized to ensure structural integrity. These results are further augmented by References 1-5, 1-6, 1-7, and 1-8 to address the effects of successive hardware improvements in Westinghouse designs, which are discussed in the following sections.

The AP1000 reactor internals are generally similar to subsequent three-loop, 12- and 14-foot core designs (specifically Doel 3 and Doel 4), which have incorporated these improvements, and on which instrumented plant test programs have been completed. The dimensions of the AP1000 core barrel wall thickness and reactor vessel to core barrel downcomer annulus are similar to those of Doel 3 and Doel 4. The AP1000 guide tube and support column designs are the same as the designs used in Doel 3 and Doel 4. The upper internals components vibration responses were measured at Doel 3 (Reference 1-9), and the lower internals were measured at Doel 4 (Reference 1-10).

The total reactor mechanical design flow rates in these instrumented tests were approximately [314,600 gpm][b] (1) and [322,500 gpm][b] as compared to the AP1000 value of [327,600 gpm][b] mechanical design flow. Table 1-1 lists the ratios of the AP1000 to Doel 4 plant velocities at various locations based on flow rates of [322,500 gpm][b] for Doel 4 and [327,600 gpm][b] for the AP1000 design. The tabulated velocities show similar values for the AP1000 and Doel 4 reactors with the exception of the inlet nozzle

ADD FOOTNOTE AS SHOWN ON ATTACHED PAGE

Footnote to Page 1-1

(1) "[ ]" indicates that the enclosed data is proprietary.
The superscripts are defined as follows:
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"b" – Information consists of supporting data relative to a process or component, structure, tool, method, etc.
"c" – Information, if used by a competitor, would reduce competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, etc. of a similar product
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AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.028 (Response Revision 1)

Question:

Reference, Volume 6, Table 3.9-5, Pg. 3.9-102:

The Level B Service loading combinations do not appear to include earthquake loading (see USNRC Standard Review Plan (SRP) 3.9.3, Appendix A, C.1.3.2). The Level C Service loading combinations do not appear to include design basis pipe break loading (see SRP 3.9.3, Appendix A, C.1.3.3). Please clarify.

NRC Follow-On Comments:

NRC does not agree that the loads from a design basis pipe break are Level C secondary loads. NRC believes that design basis pipe break loading should be included under Service Level C loading combinations in DCD Table 3.9-5.

Westinghouse Response (Revision 1):

The operating basis earthquake (OBE) has been eliminated as a design requirement for the AP1000 (see DCD section 3.7). AP1000 ASME Class 1, 2, and 3 components and structures are designed for one occurrence of the safe shutdown earthquake which is evaluated as a Service Level D condition for pressure boundary integrity (see DCD section 3.9.3.1.1). This is the same design basis as for the AP600 components and structures.

Consistent with current operating plants, Westinghouse defines the Service Level C pipe break to be a maximum of 1" in a Class 1 branch line (DCD section 3.9.1.1.3.1). This is somewhat larger than t~~T~~he design basis pipe break identified in the Standard Review Plan 3.9.3, Appendix A, Section C.1.3.3 **which is equivalent to a 3/8" break.**~~is defined as a break in Class 1 branch lines that result in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system.~~ **Breaks of 1"" and smaller do not result in dynamic mechanical loadings on reactor coolant system components and therefore are not included in DCD Table 3.9-5, which gives the loading combinations for mechanical loads.**

This pipe break does result in reactor coolant system temperature and pressure transient conditions and is thus included in the reactor coolant system design transients given in DCD Table 3.9.1. The reactor vessel internals are analyzed for each of the design transients, either individually or by using conservative enveloping transients, to show that the appropriate ASME code stress limits are met.~~The resulting loads are considered secondary loads under Service Level C conditions. Per ASME Code Section NB-3224 (Figure NB-3224-1) evaluation of these secondary loads are not required for Level C Service Limits.~~

All pipe breaks larger than 1" are Loss-of-coolant (LOCA) events **which** are considered faulted events. **The mechanical loadings resulting from a LOCA** ~~and~~ are included in DCD Table 3.9-5 under Service Level D. ~~The worst case LOCAs are considered as Level D events and envelope all the smaller LOCAs identified as emergency conditions under Level C.~~

AP1000 DCD Table 3.9-5 provides the same loading combinations as for the AP600 components and structures.

Design Control Document (DCD) Revision:

~~None~~

From DCD Revision 3, page 3.9-21:

3.9.1.1.3.1 Small Loss-of-Coolant Accident

For design transient purposes, the small loss of coolant accident is a **pipe** break equivalent to the severance of a ~~2~~**1**-inch ID branch connection ~~on the common passive core cooling system line connected~~ to the reactor **coolant system**~~vessel~~. It is assumed that the passive core cooling **system** is actuated ~~immediately after the break occurs~~ and delivers water at a minimum temperature of 70°F to the reactor vessel.

It is assumed that this transient occurs five times for design purposes.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.055 (Response Revision 1)

Question:

Current test data indicates that the ASME Code, Section III design fatigue curves may not be conservative for nuclear power plant primary system environments. The Section III Subgroup on Design (SGD) has formed a task group to provide recommendations to the SGD regarding the effect of the environment on Section III design fatigue curves. The NRC staff has been addressing the environmental fatigue issue in its review of license renewal applications. The NRC staff-referenced evaluations of the current test data are provided in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," in its license renewal reviews. Describe the method that will be used to account for the effect of the environment on the fatigue design of reactor coolant pressure boundary components in the AP-1000 plant.

Westinghouse Response:

In SECY-95-245, the NRC staff concluded that based on component sample evaluations including fatigue environmental effects, the fatigue limit would not be exceeded for most components, and that a fatigue failure of piping is not a significant contributor to core-melt frequency. Therefore no further evaluation of fatigue environmental effects on operating plants was required. The evaluations were based on typical component designs of plants with a 40 year design life. Current industry efforts to address fatigue environmental effects for license renewal are focused in the EPRI MRP ITG on Fatigue Issues. This group has proposed methods to address environmental effects in fatigue evaluations. Similar methods have been proposed and discussed by the PVRC Steering Committee on Cyclic Life and Environmental Effects, and are contained in the PVRC draft report, "Assessment of Environmental Effects on Fatigue Life in LWR Nuclear Applications", by Van der Sluys and Yakawa. These methods based on industry data will be used to evaluate the effect of environment on the fatigue design of components.

NRC Additional Comment:

Acceptance of the Westinghouse response to RAI 210.055 on the subject of environmental effects on fatigue is dependent on the industry response to staff's RAIs for the proposed EPRI methodology. Before the RAIs are resolved or the licensee decided to use the NUREG/CR reports recommended in NUREG 1801, the issue remains open for future resolution.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Westinghouse Additional Response:

The final stress analysis for the AP1000 piping is completed as part of the COL application (See DCD Section 3.9.8.2). As discussed in the original RAI response, environmental fatigue for the AP1000 piping systems will be considered in the AP1000 stress analyses consistent with the EPRI methodology. The COL applicant will address the resolution of the NRC comments on the proposed EPRI methodology as part of the evaluation of environmental fatigue.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.012 (Response Revision 1)

Question:

The application indicates that the reactor coolant pump pressure housing will be made from SA 351 or SA 352 CF3A material and that reactor coolant pressure boundary valve bodies may be castings of SA 351 CF3A. The application also indicates that cast austenitic stainless steel will not exceed a ferrite content of 30 FN (Ferrite Number). CASS reactor coolant pressure boundary components are subject to thermal embrittlement. Please provide additional information discussing the impact of this aging effect on the integrity of these components along with a discussion of how this thermal embrittlement mechanism has been considered in the design and material selection for these components. Also, please discuss the need for potential licensees of AP1000 plants to perform inspections to detect this aging effect. (Section 5.2.3)

NRC Follow-On Comment:

The Westinghouse response to this question is acceptable subject to clarification of the method used to calculate the δ-ferrite. This clarification needs to be included in the DCD.

Westinghouse Response (Revision 1):

Selection of the low carbon grade casting, CF3A, is a preventive measure considered to control the impact of thermal aging embrittlement. Also, the ferrite content of the material will be controlled by the material specifications to be below 20 FN. **Calculation of ferrite content will be based on ASTM A800 (Standard Practice for Steel Casting, Austenitic Alloy, Estimating Ferrite Content Thereof).** Based on our experience with the casting materials, there should be no significant impact of thermal aging on the integrity of the components when the carbon and ferrite content of the castings are controlled.

Inspection to detect any thermal aging effects will be part of the in-service inspection program that is the responsibility of the Combined License Applicant of each AP1000. The program will reference the edition and addenda of the ASME Code Section XI to be utilized and will be submitted to the NRC. The Combined License Applicant commitment to provide the in-service inspection program is included in DCD Section 5.2.6.2.

DCD Section 5.2.3.1 will be revised to limit the ferrite content of cast austenitic stainless steel to a maximum of 20 FN.

DCD Section 5.2.3.1 will be revised to include the standard used to calculate ferrite content.

Design Control Document (DCD) Revision:

From DCD page 5.2-9, Section 5.2.3.1: (Already Incorporated in DCD Revision 3)

Table 5.2-1 material specifications are the materials used in the AP1000 reactor coolant pressure boundary. The materials used in the reactor coolant pressure boundary conform to the applicable ASME Code rules. Cast austenitic stainless steel does not exceed a ferrite content of ~~30~~ 20 FN.

From DCD Revision 3 page 5.2-9, Section 5.2.3.1:

Table 5.2-1 material specifications are the materials used in the AP1000 reactor coolant pressure boundary. The materials used in the reactor coolant pressure boundary conform to the applicable ASME Code rules. Cast austenitic stainless steel does not exceed a ferrite content of 20 FN. **Calculation of ferrite content is based on ASTM A800.**

From DCD Revision 3 page 5.2-30:

5.2.7 References

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

2. EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report, Interim Report, April, 1982.

3. Logsdon, W. A., Begley, J. A., and Gottshall, C. L., "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metals," WCAP-9292, March 1978.

4. Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary), March 1970, and WCAP-7735 (Nonproprietary), August 1971.

5. Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1975.

6. Enrietto, J. F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976.

7. **ASTM A800, "“Standard Practice for Steel Casting, Austenitic Alloy, Estimating Ferrite Content Thereof”".**

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.021 (Response Revision 1)

Question:

In the AP600 review, RAIs 251.2 through 251.23 pertain to RCP flywheel integrity. In addition, WCAPs-13734 and 13735, "Structural Analysis Summary for the AP600 Reactor Coolant Pump Flywheel," were submitted as supplemental information for the revised response to question 251.11. Confirm that these responses and the WCAPs are applicable to the AP1000 application as it pertains to RCP flywheel integrity. Should aspects of these responses or reports not be applicable, provide updated information to address the AP600 RAIs as applicable to AP1000 RCP flywheel integrity. (Section 5.4.1)

Note: AP600 RAIs 251.2 through 251.23 were issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its responses to these RAIs in letters dated January 14, May 24, and May 28, 1993 (NUDOCS Accession Nos. 9301250260, 9306020387, and 9306020220, respectively).

NRC Follow-On Comments:

In response to AP600 RAI 251.3, the applicant states that the critical flaw size for the AP1000 flywheel design was approximately 1 inch, as opposed to 2 inches critical flaw size that was reported in response to the RAI for the AP600 designs. However WCAP-15994-P, Revision 0, indicates that the critical flaw size for the AP1000 and AP600 designs are virtually the same (1.075 inches for AP1000 vs 1.16 inches for the AP600 design). There appears to be an error in the AP 600 RAI response. The applicant needs to clarify what the actual critical flaw sizes are for the two flywheel designs.

In response to AP600 RAI 251.8, the applicant's response for the AP1000 flywheel design is virtually the same as that for the AP600 flywheel design, with the exception that the response for the AP1000 increases the kinetic energy for flywheel fragments that could potential impact the RCP structure from 10% of the tensile energy-absorbing capability in AP600 RCP structure to 15% of the tensile energy-absorbing capability in AP1000 RCP structure. However, there is some confusion in the wording of the final paragraph of the applicant's AP1000 response to AP600 RAI 251.8 as to whether potential flywheel fragments will penetrate the flywheel enclosure or not, and whether they could impact the surrounding RCP structure.

In response to AP600 RAIs 251.17 and 251.19, the applicant discusses the relationship of the structural integrity of the RCP flywheel enclosures to revolution of the flywheels at normal and design operating speeds. In this case, the applicant's responses for AP1000 are virtually the same as the previous AP 600 design responses to the RAIs. However, in these AP1000 responses to these AP600 RAIs, the applicant has a statement that the flywheel impact on the

enclosure was assessed for a postulated rupture of the AP600 flywheel under normal and design revolution speeds and that the impact on the AP600 enclosure welds was within acceptable ASME limits. The applicant then concludes that since the AP1000 flywheel design is similar to that for the AP600 flywheels, "it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits" during revolutions of the AP1000 flywheel at normal speeds (as addressed in the AP1000 response to AP600 RAI 251.17) and design speeds (as addressed in the AP1000 response to AP600 RAI 251.19). The applicant needs to state whether this is based solely on engineering judgement or whether the applicant actually did the corresponding analyses at normal speed and design speed for the AP1000 design.

Westinghouse Response (Revision 1):

Responses to AP600 RAIs 251.2 through 251.23 specific to the AP1000 design are given below. The format is to repeat the AP600 question and provide a response specific to the AP1000 design. WCAP-15994-P, **Revision 1** (Proprietary), WCAP-15994-NP, **Revision 1** (Non-Proprietary), "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel", November 2002, has been issued to supplement some of the responses given below. In the responses below, this WCAP is referred to as Reference 1.

When the responses below refer to other AP600 RAI responses, the reference is to the AP1000 response to the AP600 RAI as given here in the overall response to this AP1000 RAI.

AP600 RAI 251.2
Westinghouse proposes to use a depleted uranium alloy casting in an Inconel alloy welded enclosure to construct the pump flywheel. These materials are not addressed in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity". Provide technical justifications for the use of these materials (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.2
As noted in Subsection 5.4.1 of the DCD, the AP1000 canned motor reactor coolant pump uses a fundamentally different approach to demonstrate safe operation of the flywheel than the design approach for which Section 5.4.1.1 of the Standard Review Plan and Regulatory Guide 1.14 were developed. Of prime importance in the consideration of flywheel integrity is minimizing the potential for generation of missiles from the flywheel in conformance with the requirements of General Design Criteria 4. The AP1000 approach is to demonstrate that fragments from a postulated flywheel fracture do not penetrate the surrounding pressure boundary and thus do not become missiles. See the AP1000 response to AP600 RAI 251.11 for additional information on the analysis of the retention of flywheel fragments. This basis of containing postulated fragments is the same as for the rotor and other rotating parts in previous shaft seal pump designs. The approach behind the recommendations of Section 5.4.1.1 of the Standard Review Plan and Regulatory Guide 1.14 is to minimize the potential for a flywheel fracture by extensive testing and inspection of the flywheel.

Although conformance with the recommendations in the regulatory guide is not the only means to demonstrate safe operation of the pump, many of the recommendations are followed in the design and fabrication of the flywheel assembly for operational reliability. Since the AP1000 design does not rely on flywheel material integrity to minimize the potential for the generation of missiles, the quality assurance requirements inherent in the use of ASME Code pressure boundary quality material as suggested by the Standard Review Plan are not required. The design requirements for the flywheel assembly materials are selected to provide a high level of operational reliability. The basis for the design requirements for the flywheel assembly materials is outlined below.

The flywheel assembly is a uranium-alloy casting or forging surrounded by a nickel-chromium-iron alloy enclosure. The material strength used for the analyses that demonstrate flywheel integrity is based on the material specification outlined in AP1000 DCD Table 5.4-2. The material toughness is demonstrated by the yield strength and elongation. See the AP1000 response to AP600 RAI 251.3 for additional information on the fracture toughness properties of the uranium alloy. Since the uranium alloys to be used in the flywheel were not developed for use as pressure boundary materials, ASME Code material specifications do not exist. See the AP1000 response to AP600 RAI 251.23 for additional information on the material specification. Nevertheless, quality assurance practices can confirm that the minimum material requirements are met. The nickel-chromium-iron Alloy 690 material used in the enclosure is a commonly used material with established material specifications.

The uranium alloy does not come in contact with the reactor coolant. The Alloy 690 enclosure material has been shown to be compatible with reactor coolant in other applications. The operating temperature of the coolant surrounding the flywheel assembly is substantially less than the reactor coolant system operating temperature, so stress corrosion cracking of the Alloy 690 is not expected to be an issue. See the AP1000 response to AP600 RAI 251.21 for additional information on the resistance to stress corrosion cracking of the flywheel enclosure.

AP600 RAI 251.3

Westinghouse indicates that the fracture toughness guidelines in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14 are not applicable to depleted uranium alloy castings. Provide information on the fracture toughness properties for this material and propose fracture toughness requirements with technical justifications (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.3

The fracture toughness of the uranium alloy casting is approximately 50 ksi √in. between 100°F and 200°F based on available data. Over the same temperature range the minimum impact energy (Charpy V-notch) is 10 foot-pounds. The material specification for the flywheel material includes a requirement for this minimum impact energy. The material specification does not include a fracture toughness requirement, but the properties and processing specified define a material that meets the 50 ksi √in. minimum.

Calculation of the critical flaw sizes (Reference 1) is based on the 50 ksi √in. fracture toughness. The minimum critical flaw size is greater than 1 inch for a full-length axial crack on the inner diameter. This flaw size was calculated for assembly plus design conditions (125% overspeed).

The minimum critical flaw sizes for a full-length axial crack reported in WCAP-15994, Revision 0 for both the AP600 (1.16 inches) and AP1000 (1.075 inches) flywheels are correct. The critical flaw size for a semi-elliptical (critical radius) crack is greater than 2 inches for both the AP600 and AP1000 flywheels.

AP600 RAI 251.4
Provide information on the fabrication process and resulting quality for the depleted uranium alloy casting (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.4
The melting of the depleted uranium alloy for the flywheel casting or forging billet is done under vacuum or inert atmosphere to provide a high quality product. The vacuum or inert atmosphere prevents reaction of the uranium with air and minimizes the potential for the formation of voids. Because of the density of the uranium alloy, slag and other impurities tend to float to the top of the molten metal and porosity in the cast material is not a problem. The molds for the casting are treated to minimize the contamination of the uranium with carbon. The rest of the manufacturing process is controlled to minimize the contamination of the uranium alloy with carbon and hydrogen. Excessive carbon reduces the ductility of the uranium alloy. Hydrogen contamination may induce delayed cracking. Because of the thickness of the flywheel, the final heat treatment is a solution anneal in a vacuum furnace followed by a slow cooling. Other heat treatments such as annealing followed by water quenching and aging hardening are not appropriate for a thick uranium alloy flywheel. See the AP1000 response to AP600 RAI 251.22 for additional discussion of the heat treatment.

AP600 RAI 251.5
Section 1A of the SSAR indicates that the AP600 design meets the guidelines of Regulatory Position 1.d in Regulatory Guide 1.14. However, the flywheel, including the enclosure welds, will not be inspected. Discuss how the flywheel design meets Regulatory Position 1.d.

Westinghouse AP1000 Response to AP600 RAI 251.5

The uranium alloy flywheel is not subject to welding operations, including repair welding, or any other finishing operations that use thermal methods. The component parts of the enclosure are connected together with flexible, full-penetration welds. These welds are inspected following fabrication by ultrasonic testing and liquid penetrant testing. ASME Code, Section III criteria for structural welds are used as guidelines to establish welding and inspection requirements. See the AP1000 response to AP600 RAI 251.14 for additional information on the analysis and inspection of the enclosure flexible welds. The enclosure represents only a small fraction of the energy in a rotating flywheel assembly. The locations of the flexible welds are such that there is minimal effect on the fracture analysis.

AP600 RAI 251.6
Regulatory Positions 2.c, 2.d, and 2.e in Regulatory Guide 1.14 recommends that an analysis be submitted for staff review. Provide the analysis with appropriate technical justifications. Further, because no inservice inspection for the flywheel is being proposed, describe the flaw size assumed in its analysis (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.6
Regulatory Positions 2.c., 2.d., and 2.e. in Regulatory Guide 1.14 recommend that analyses be conducted to predict the critical speed for ductile failure, nonductile failure, and excessive deformation of the reactor coolant pump flywheel. As noted in Subsection 5.4.1 of the DCD and the AP1000 response to AP600 RAI 251.2, the approach to demonstrate safe operation of the AP1000 canned motor reactor coolant pump flywheel differs from the design approach for which Regulatory Guide 1.14 was developed. The AP1000 design approach of demonstrating that postulated flywheel fragments are contained by the pump structure limits the significance of the analysis of critical flywheel failure speeds.

The analysis completed for the flywheel structure evaluates the stress intensity levels at the normal speed and at the design speed of 125 percent of normal. The calculated stress levels are evaluated against ASME Code, Section III, Subsection NG stress limits and the recommended stress limits in Positions 4.a. and 4.c. of Standard Review Plan 5.4.1.1. of one-third and two-thirds of yield stress for normal speed and design speed, respectively. The margin inherent in these limits provides an appropriate degree of margin to failure at the normal and design speeds. See Reference 1 for additional information on the evaluation of stress in the flywheel assembly.

The flaw size assumed in the evaluation of fracture toughness is described in the AP1000 response to AP600 RAI 251.3.

AP600 RAI 251.7
Section 1A of the SSAR indicates conformance with Regulatory Position 2.f in Regulatory Guide 1.14. Provide information to support this statement.

Westinghouse AP1000 Response to AP600 RAI 251.7
As noted in Subsection 5.4.1.3.6.3 of the AP1000 DCD and in the AP1000 response to AP600 RAI 251.8, the design speed (125 per cent of normal speed) envelopes all expected and postulated overspeed conditions including overspeeds due to postulated pipe ruptures. See the AP1000 response to AP600 RAI 251.8 for a discussion of the size of postulated pipe ruptures also. This limitation on the potential overspeed along with the design approach of demonstrating that postulated flywheel fragments are contained by the pump structure limits the significance of the analysis of critical flywheel failure speeds. The analysis completed for the flywheel stress report evaluates the stress intensity levels at the normal speed and the design speed of 125 percent of normal. The calculated stress levels satisfy the ASME Code, Section III, Subsection

NG stress limits. The calculated primary stress levels are less than the recommended stress limits in Positions 4.a. and 4.c. of Standard Review Plan 5.4.1.1 of one-third and two-thirds of yield stress for normal speed and design speed, respectively. See Reference 1 and the AP1000 responses to AP600 RAIs 251.16, 251.17, 251.18, and 251.19 for additional information on the evaluation of stress in the flywheel assembly.

The flywheel structural analysis verifies that the failure modes outlined in Positions 2.c, 2.d, and 2.e of Regulatory Guide 1.14 do not occur at the design speed. The flywheel stress evaluation noted above demonstrates an appropriate margin against these failure modes. In addition, the design of the canned motor pump mitigates the effects of hypothetical failures by these modes, as outlined below.

The AP1000 response to AP600 RAI 251.11 discusses the containment of fragments from a postulated flywheel fracture. The mode of failure, ductile or nonductile, would not alter the capacity of the surrounding pump structure to absorb the energy of the fragments and prevent the generation of missiles from the flywheel assembly.

Regulatory Guide 1.14 defines excessive deformation as any deformation that could cause separation of the flywheel from the shaft. Because of the restriction of the lateral movement of the flywheel assembly by the surrounding structure and axial movement by the thrust bearings, the loss of shrink fit would not be expected to result in substantial movement of the flywheel assembly or significant separation of the assembly from the shaft. This restriction in movement of the flywheel assembly and the adjacent location of the journal bearing to the flywheel assembly minimize the potential for a structural failure of the shaft during a hypothetical overspeed transient sufficient to result in excessive deformation.

Neither separation of the flywheel assembly from the shaft nor structural failure of the shaft would result in a loss of safety-related function of the canned motor pump during an overspeed transient. That safety-related function is the maintenance of the primary pressure boundary. Neither separation of the flywheel assembly nor structural failure of the shaft would degrade the pressure boundary of the pump. The safety-related function of providing flow during coastdown of the pump is not germane during an overspeed transient.

AP600 RAI 251.8

Section 1A of the SSAR indicates conformance with Regulatory Position 2.g in Regulatory Guide 1.14, relating to the flywheel overspeed due to postulated pipe rupture. Section 5.4.1.3.6.3 of the SSAR appears to assume the application of leak-before-break (LBB) for all high-energy piping 10 cm (4 in) in diameter or larger. Since the outcome of the staff's review of the application of LBB to the AP600 design is uncertain, the staff recommends that Westinghouse discuss how the flywheel conforms with RG 1.14 if the criteria of Section 3.6.2 and BTP MEB 3-I is used to determine pipe break size.

Westinghouse AP1000 Response to AP600 RAI 251.8

As was the case for AP600, for AP1000 nominal pipe sizes of 6" and larger are qualified for elimination of post-rupture dynamic analysis through application of leak-before-break criteria.

Therefore, the largest break analyzed to determine the dynamic response of the AP1000 reactor coolant pump is that of a 4" pipe (e.g. pressurizer spray line, first stage ADS line).

The overspeed analysis of the AP1000 reactor coolant pump flywheel is based on the design speed of 125 per cent of normal speed. The AP1000 pipe rupture overspeed is expected to be enveloped by the design speed since the reactor coolant main loops and all of the branch line piping with a nominal diameter of 6 inches and greater are being qualified for LBB. The pipe rupture overspeed is expected to be substantially less than any of the calculated critical flywheel failure speeds.

As noted in the AP1000 response to AP600 RAI 251.2, the approach used to demonstrate the safe operation of the flywheel is containment of the fragments from a postulated fracture by the surrounding pump ~~structure~~ **pressure boundary. In the evaluation of the postulated flywheel fracture, the flywheel enclosure is conservatively neglected. This approach maximizes the energy of the flywheel fragments that are assumed to impact the pump pressure boundary.** For a postulated flywheel fracture at the flywheel design speed there is a large amount of margin in the calculated capability of the pump ~~structure~~ **pressure boundary** to contain flywheel fragments. **In the AP600 pump design, the flywheel fragments resulting from a fracture had only 8.5% of the energy required to penetrate the pump pressure boundary. Although the energy of the flywheel fragments in the AP1000 design has increased to 12% of the energy required to penetrate the pump pressure boundary, the conclusion that the fragments from a postulated flywheel fracture will not penetrate the pressure boundary is still valid.** Thus even in the event of a postulated failure of a flywheel during a hypothetical break of a reactor coolant loop pipe, it is not expected that additional breaks in the reactor coolant pressure boundary would be created nor would missiles be generated by the flywheel.

AP600 RAI 251.9

Section 1A of the SSAR indicates that Westinghouse is taking exception to Regulatory Position 4.a in Regulatory Guide 1.14. Propose an alternative to this position with appropriate technical justifications.

Westinghouse AP1000 Response to AP600 RAI 251.9

A spin test is done on the flywheel assembly after the enclosure is welded closed. Inspection of the flywheel inside the assembly is not practical. Because of the density of the uranium, radiographic examination is also not a practical option.

The uranium alloy flywheel is ultrasonically inspected following final machining and prior to assembly of the enclosure around the flywheel. The ultrasonic inspection conforms to the requirements of the ASME Code, Section III, paragraph NB-2574, for ferritic steel castings, including the use of the procedures outlined in SA-609 (ASTM-A-609). See the AP1000 response to AP600 RAI 251.13. Machined surfaces of the uranium flywheel undergo liquid penetrant inspection prior to final assembly. The liquid penetrant inspection conforms with the requirements of the ASME Code, Section III, paragraph NB-2576, including the use of the procedures outlined in SA-165 (ASTM-A-165).

In-process controls during the assembly of the enclosure onto the flywheel are used to provide for the quality of the completed assembly. The spin test of the completed assembly confirms the quality of the flywheel assembly. Since the basis for safe operation of the flywheel assembly is the retention of the fragments from a postulated fracture by the structure of the pump, inspection subsequent to the spin test is not necessary for safe pump operation.

AP600 RAI 251.10

Performance of inservice inspection of the flywheel should be considered. If the ISI procedures in Section 5.4.1.1 of the SRP are not applicable to uranium flywheels, propose alternative inservice inspection procedures with appropriate technical justifications (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.10

Inservice inspection of the uranium alloy flywheel would be very labor intensive and involve significant radiation exposure. Since the surrounding structure of the pump would contain flywheel fragments even in the worse case fracture, inservice inspection would do little to increase the safety of pump operation. The technical justification of no inservice inspection is the analysis that shows that the fragments of a fractured flywheel would not penetrate the pressure boundary of the pump to become missiles (see the AP1000 response to AP600 RAI 251.11). On this basis a flywheel fracture is an operational reliability consideration rather than a safety-related consideration. The use of inspections and in-process controls during fabrication of the flywheel assembly and a spin test of the completed assembly also provide verification of the initial quality of the assembly. The use of vibration monitoring of the pump during operation provides an indication of rotating part stability and thus integrity. This allows any necessary maintenance to be performed as needed for operational reliability.

As noted in the AP1000 response to AP600 RAI 251.2 the design approach to the flywheel in the AP1000 canned motor reactor coolant pump is fundamentally different than that for previous shaft seal reactor coolant pump designs. The canned motor pump design was selected for several safety related and operational reasons. Inherent in the design of a canned motor reactor coolant pump is the location of the flywheel assembly within a pressure housing and the flywheel enclosure in contact with reactor coolant. To make the flywheel readily accessible for an inservice inspection of marginal utility, many advantages of the canned motor pump would have to be foregone. Routine inservice inspection of the flywheel is neither recommended nor advantageous.

AP600 RAI 251.11

Section 1A of the SSAR states that a flywheel rupture will be contained within the stator shell. Provide an analysis and technical justifications supporting this statement.

Westinghouse AP1000 Response to AP600 RAI 251.11

The canned motor reactor coolant pump has an outer shell that comprises the pressure boundary. The shell is analyzed to demonstrate that in the event of a postulated flywheel fracture, the surrounding pump structure is sufficient to prevent missiles from leaving the pump. The analysis considers that portion of the shell, including the flange, and motor end cap around the flywheel assembly between the top and bottom elevations of the assembly as the barrier to

missile generation. The structural analysis summary is documented in Reference 1 and is outlined below.

The analysis of the capacity of the surrounding pump structure to contain the fragments of a postulated flywheel failure is done using the energy absorption equations of Hagg and Sankey (Hagg, A. C., and Sankey, G. O., "The Containment of Disk Burst and Fragments by Cylindrical Shells," ASME Journal for Power, April 1974, pp. 114-123). The containment of missile-like metal disk fragments is by a two-stage process. Stage 1 involves inelastic impact and transfer of momentum to include an effective target mass. To show that the fragments do not perforate the surrounding structure, the energy dissipated in plastic compression and shear strain and the local impact area must be sufficient to account for the loss in kinetic energy of the system. For the nonperforation case the process enters Stage 2, which involves dissipation of energy in plastic tension strain over extended volumes of shell material. For containment, the energy dissipated in plastic strain in Stage 2 must account for the residual kinetic energy on the system. In predictive calculations it is more conservative to consider Stage 2.

For the AP1000 reactor coolant pump analysis, the uranium insert in the flywheel assembly is assumed to fracture at the design speed of 125 percent of normal speed. The worst-case scenario of fragment size and number was derived analytically, using methods from Hagg and Sankey to determine the mass and velocity combination that would produce the most severe impact on the surrounding pressure boundary components. The following conservative assumptions are also made:

1. End plates and welds of the flywheel enclosure and the coolant surrounding the flywheel assembly have negligible energy-absorbing capability.
2. Only the mass in the stator shell and flange and the motor end cap between the elevation of the top and bottom of the flywheel assembly are considered to absorb energy.
3. Closure bolts and joint effects were not considered to be affected.
4. The minimum material properties were used.

The analysis results show that the fragments impact the surrounding pump structure with a kinetic energy of less than 15 percent of the tensile energy-absorbing capability of the surrounding pump structure. Thus the components around the flywheel contain the flywheel fragments using only a small portion of the energy-absorbing capability. The energy absorbed by the flywheel enclosure is small compared to the surrounding pump structure and was not considered in the calculation of flywheel fragment containment within the pump pressure boundary.

AP600 RAI 251.12

Section 1A of the SSAR indicates that a "small" flywheel rupture or leak in the enclosure will not result in stresses in the pressure boundary to cause a break. Provide information to clarify what is the intent of the term "small" flywheel rupture. The staff is concerned with the rupture of the flywheel into large fragments of high energy.

Westinghouse AP1000 Response to AP600 RAI 251.12
The canned motor pump design is evaluated for a spectrum of postulated uranium flywheel fractures. A fracture that ruptures the flywheel enclosure is bounded by the analysis of the worst-case fracture (Reference 1) that shows that the fragments are contained as noted in AP1000 DCD section 5.4.1.3.6.3. A fracture that deforms the enclosure enough to bring it in contact with the surrounding structure is bounded by the analysis described in AP1000 DCD section 5.4.1.3.6.2. A small fracture in the context of the DCD discussion is one that may unbalance the assembly, but any resulting fragment is contained by the enclosure without sufficient deformation to result in interference with the surrounding structure. The discussion of these faults on the low end of the spectrum are included for completeness of the discussion of postulated flywheel fractures.

AP600 RAI 251.13
Section 5.4.1.3.6.3 of the SSAR indicates that ultrasonic inspection of the uranium following final machining will be based on ASTM A388 as modified for uranium. Identify any modifications to the application of ASTM A388 to the AP600 design with appropriate technical justifications. In addition, demonstrate that this preservice inspection is equivalent to that in Section III of the ASME Code.

Westinghouse AP1000 Response to AP600 RAI 251.13
ASTM A388, which is a standard for use of ultrasonic inspections on steel forgings, is not given as the standard for ultrasonic inspection of the uranium following final machining in AP1000 DCD section 5.4.1.3.6.3. ASTM A609, which is a standard for use of ultrasonic inspections on ferritic steel castings, will be used as the standard for ultrasonic inspection of the uranium flywheel. Changes to the practices specified in the standard to account for use on uranium include the use of uranium reference blocks and potential additional restrictions on the couplants used. The size and frequency of transducers may also be different than the standard, although the inspection of a prototype flywheel casting was done with a transducer size and frequency in the range designated in the standard. Areas of the standard that are not dependent on the type of material inspected, such as personnel qualification requirements, surface conditions, procedures, and data reporting should not have to be modified. See the AP1000 response for AP600 RAI 251.9 for additional discussion of inspection of the uranium flywheel.

It is not the intent that the inspection of the uranium alloy flywheel be equivalent in every respect to inspections required of components built to the requirements of the ASME Code, Section III. The requirements for the flywheel are chosen to provide high operational reliability. There are no pressure boundary functions associated with the flywheel assembly that require the use of the ASME Code.

AP600 RAI 251.14
Demonstrate that the construction of the flywheel enclosure meets Section III of the ASME Code, including inspection (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.14
Since the flywheel enclosure is not a pressure boundary and is not relied upon to contain fragments from a postulated flywheel fracture, there is no requirement to meet the requirements of the ASME Code, Section III for construction of the enclosure. Additionally, the enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The function of the enclosure is to isolate the uranium alloy from the reactor coolant circulating in the reactor coolant pump. A leak in the enclosure could result in an out-of-balance condition for the flywheel assembly or, over the long term, the possible introduction of depleted uranium into the reactor coolant. Neither of these events represents a catastrophic failure and both would be addressed by other systems. Sensors in the pump detect vibration of the pump and the chemical and volume control system includes provisions to reduce contaminants in the reactor coolant. The uranium would be detected by periodic sampling of the reactor coolant by the primary sampling system.

The ASME Code, Section III criteria for structural welds are used to establish welding requirements and inspection requirements for the enclosure. As noted in the AP1000 DCD section 5.4.1.3.6.3, the welds are subject to dye penetrant and ultrasonic tests. The ASME Code Subsection NG stress limit criteria are used as guidelines to evaluate the stress in the enclosure components and the flexible welds for normal and design speeds. The use of the ASME Code, Section III to establish design, fabrication, and inspection requirements was selected to provide operational reliability and availability.

AP600 RAI 251.15
Demonstrate that the design overspeed of the flywheel is at least 10% above the highest anticipated overspeed (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.15
The requirement for the AP1000 is that the design speed (125 percent of normal speed) be greater than or equal to anticipated overspeed conditions due to electrical faults and overspeed conditions due to postulated pipe breaks. Anticipated overspeed conditions are those due to electrical faults including turbine overspeed events. Because of design of the turbine control system (see AP1000 DCD section 10.2.2), reactor coolant pump overspeed resulting from an electrical fault is expected to be less than the design speed. See the AP1000 response to AP600 RAI 251.8 for a discussion of flywheel overspeed due to postulated pipe rupture.

Since the basis for safe operation of the pump with respect to flywheel integrity is the containment of flywheel fragments by the pump structure rather than the prevention of fracture (see the AP1000 responses to AP600 RAIs 251.2 and 251.11), a 10% margin between calculated overspeed and the design speed is not necessary to assure safe operation.

AP600 RAI 251.16
Show that the combined stresses for the uranium flywheel at the normal operating speed, due to centrifugal forces and the interference fit of the wheel on the shaft, is less than 1/3 of the minimum specified yield strength (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.16
The flywheel structural analysis verifies that the primary stresses in the uranium due to centrifugal forces at the normal operating speed are less than one-third of the minimum yield strength. The combination of primary and secondary stresses is evaluated using stress limits in the ASME Code, Section III, Subsection NG. The secondary stresses are due to the interference fit of the uranium on the shaft. The allowable stress values developed applying ASME Code, Section III, factors (Appendix III) to the mechanical properties of uranium are satisfied for analyzed stresses at normal operating speed. See Reference 1 for analysis details.

AP600 RAI 251.17
Discuss how the limit in AP600 RAI 251.16 is met for the flywheel enclosure and associated welds (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.17
The evaluation of the flywheel enclosure does not use the limit of one-third of minimum yield strength as a criterion for normal operating speed. The flywheel enclosure **only** prevents contact of coolant with the uranium flywheel**, and is not considered to be a "reactor coolant pressure boundary"**. No credit is taken in the analysis of the flywheel missile generation for the retention of the fragments by the enclosure, and the flywheel enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The evaluation of the stress in the flywheel enclosure components and the flexible welds connecting the components for normal and design speeds uses the criteria in Subsection NG of the ASME Code as a guideline. The ASME Code limits are satisfied for analyzed stresses of the flywheel enclosure at the normal operating speed (see Reference 1).

The AP600 and AP1000 flywheel enclosure designs are very similar. **The radial thickness of the two enclosures is approximately the same.** The stresses **calculated** in the AP1000 flywheel enclosure **shell components** at normal operating speed are similar to those in the AP600 flywheel enclosure. The maximum radial displacement ~~at~~ **of** the AP1000 flywheel enclosure welds is also **very** similar to that for the AP600 flywheel enclosure welds. The AP600 flywheel enclosure welds ~~were~~ **have been** shown to meet the ASME Code limits during operation at the normal operating speed. Therefore, **because of the similarities in enclosure configuration, operating conditions, enclosure shell stresses, radial thickness, and maximum weld area radial displacement between the AP600 and AP1000,** it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits during normal operating conditions. **Stress analyses of the flywheel enclosure welds under normal operating speed conditions will be performed during detailed design of the AP1000 reactor coolant pump.**

A discussion of the flywheel enclosure weld stresses will be included in Revision 1 of Reference 1.

AP600 RAI 251.18
Show that the combined stresses for the uranium flywheel at the design overspeed, due to centrifugal forces and the interference fit, is less than 2/3 of the minimum specified yield strength (Section 5.4.1).

Westinghouse AP1000 Response to AP600 251.18
The flywheel structural analysis verifies that the combined stresses in the uranium flywheel due to centrifugal forces and the interference fit at the design speed of 125 percent of normal speed are less than the limit of two-thirds of the minimum yield strength. The combination of primary and secondary stresses is also evaluated using stress limits in the ASME Code, Section III, Subsection NG. The secondary stresses are due to the interference fit of the uranium on the shaft. The allowable stress values developed applying ASME Code, Section III, factors (Appendix III) to the mechanical properties of uranium are satisfied for analyzed stresses at the design speed (see Reference 1).

AP600 RAI 251.19
Discuss how the limit in AP600 RAI 251.18 is met for the flywheel enclosure and associated welds (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.19
The evaluation of the flywheel enclosure does not use the limit of two-thirds of minimum yield strength as a criterion for design speed conditions. The criteria in the ASME Code, Section III, Subsection NG, are used as a guideline for stress limits. The ASME Code limits are satisfied for analyzed stresses in the flywheel enclosure at the design speed. See Reference 1 and the AP1000 response for AP600 RAI 251.17.

The AP600 and AP1000 flywheel enclosure designs are very similar. **The radial thickness of the two enclosures is approximately the same.** The stresses **calculated** in the AP1000 flywheel enclosure **shell components** at design speed are similar to those in the AP600 flywheel enclosure. The maximum radial displacement ~~at~~ **of** the AP1000 flywheel enclosure welds is also **very** similar to that for the AP600 flywheel enclosure welds. The AP600 flywheel enclosure welds ~~were~~ **have been** shown to meet the ASME Code limits during operation at the design speed. Therefore, **because of the similarities in enclosure configuration, operating conditions, enclosure shell stresses, radial thickness, and maximum weld area radial displacement,** it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits during operation at the design speed. **Stress analyses of the flywheel enclosure welds under design speed conditions will be performed during detailed design of the AP1000 reactor coolant pump.**

A discussion of the flywheel enclosure weld stresses will be included in Revision 1 of Reference 1.

AP600 RAI 251.20
Demonstrate that the shaft and the bearings supporting the flywheel will be able to withstand any combination of loads from normal operation, anticipated transients, the design basis loss-of-coolant accident, and the safe shutdown earthquake (Section 54.1).

Westinghouse AP1000 Response to AP600 RAI 251.20
The containment of fragments from a postulated fracture of the flywheel is not dependent on the support of the shaft and flywheel by the bearings. Postulated failures of the bearings and shaft would result in the rotating assembly being slowed to a stop. Bearing or shaft failures would be indicated by vibration or temperature sensors. A postulated failure of a bearing or shaft that allowed excessive lateral movement would result in contact between one or more rotating parts and the surrounding structure thereby slowing the rotation. A postulated failure of a bearing or shaft that allowed excessive axial movement would not remove the restriction provided by the pump internals, including the impeller and suction adapter. Thus a failure that would allow axial movement would not result in significant movement of the flywheel assembly.

Based on this information, the effect of these loads on the shaft and bearings is of interest with regard to operational reliability but not with regard to safe operation. The shaft and bearing supports are evaluated for loads due to seismic events.

AP600 RAI 251.21
Identify the materials for the flywheel enclosure and associated welds. Provide technical justification to show that the flywheel enclosure and associated welds are resistant to stress corrosion cracking, especially if Inconel 600 or 182 materials will be used (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.21
The material of construction of the flywheel assembly enclosure is nickel-chromium-iron Alloy 690. The material for the welding filler metal is nickel-chromium-iron Alloy 52. Since the coolant surrounding the flywheel assemblies is normally at a relatively low temperature (approximately 165 F) and Alloy 690 has shown good resistance to stress corrosion cracking in applications at the higher reactor coolant system temperatures, primary water stress corrosion cracking in the flywheel assembly would not be expected.

AP600 RAI 251.22
Demonstrate that the uranium flywheel is resistant to stress corrosion cracking or other potential degradation mechanisms in a reactor coolant environment (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.22
The uranium alloy flywheel is sealed in the nickel-chromium-iron alloy enclosure and is not in contact with the reactor coolant or other fluid. See the AP1000 response to AP600 RAI 251.14 for additional discussion of the enclosure flexible welds. The uranium alloy flywheel is heat

treated by solution annealing in a vacuum furnace and slowly cooled. This heat treatment minimizes the potential for residual stresses. The heat treatment process also removes hydrogen from the material to reduce the potential for hydrogen embrittlement. Since the depleted uranium alloy is not in contact with reactor coolant or any other fluid and operates at a relatively low temperature, degradation of the material is not expected.

AP600 RAI 251.23

Table 5.4-2 in the SSAR lists the flywheel material specifications. Provide the technical basis for these specifications.

Westinghouse AP1000 Response to AP600 RAI 251.23

The material specification information including ultimate tensile strength and yield strength provided in AP1000 DCD Table 5.4-2 is based on material testing by the material supplier. The composition of the alloys, including the limits on the constituent elements, is also based on the experience of the material supplier. The production of the uranium flywheel is controlled to minimize the formation of voids or other defects. The heat treatment process is controlled to provide the required material properties. See the AP1000 response to AP600 RAI 251.22 for a discussion of the heat treatment. Quality assurance testing of the material verifies that the material supplied conforms to the material specification. Ultrasonic and liquid penetrant inspections are performed on the uranium flywheel to verify the absence of unacceptable defects. See the AP1000 responses to AP600 RAIs 251.9 and 251.13 for a discussion of the ultrasonic and liquid penetrant inspections.

Design Control Document (DCD) Revision:

See the response to AP1000 RAI 440.040. (These changes were incorporated into DCD Revision 3.)

From DCD Revision 3 page 1.6-12:

Table 1.6-1 (Sheet 11 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
5.2	WCAP-8324-A	Control of Delta Ferrite in Austenitic Stainless Steel Weldments, June 1975
	WCAP-8693	Delta Ferrite in Production Austenitic Stainless Steel Weldments, January 1976

Table 1.6-1 (Sheet 11 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
5.3	WCAP-15557	Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology, August 2000
5.4	**WCAP-15994-P (P)** **WCAP-15994-NP**	**Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel**
6.2	WCAP-8077 (P) WCAP-8078	Ice Condenser Containment Pressure Transient Analysis Methods, March 1973
	WCAP-8264-P-A (P) WCAP-8312-A	Westinghouse Mass and Energy Release Data for Containment Design, June 1975 (P), August 1975 (Non-P)
	WCAP-10325 (P)	Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version, May 1983
	WCAP-8822 (P) WCAP-8860	Mass and Energy Releases Following A Steam Line Rupture, September 1976
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, June 1984
	WCAP-14234	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, June 1997
	WCAP-15846 (P) WCAP-15862	WGOTHIC Application to AP1000, Revision 0, April 2002

(P) Denotes Document is Proprietary

From DCD Revision 3 page 5.4-6:

5.4.1.3.6.3 Flywheel Integrity

The canned-motor reactor coolant pump in the AP1000 complies with the requirement of General Design Criterion (GDC) Number 4. That Criterion states that components important to safety be protected against the effects of missiles.

The flywheel assemblies are located within and surrounded by the heavy walls of the motor end closure, casing, thermal barrier flange, stator shell, or main flange. In the event of a postulated worst-case flywheel assembly failure, the surrounding structure can, by a large margin, contain the energy of the fragments without causing a rupture of the pressure boundary. The analysis **in Reference 10** of the capacity of the housing to contain the fragments of the flywheel is done using the energy absorption equations of Hagg and Sankey (Reference 2).

From DCD Revision 3 page 5.4-76:

5.4.16 References

1. Eshleman, R. L., "Flexible Rotor-Bearing System Dynamics, Part I. Critical Speeds and Response of Flexible Rotor Systems," Flexible Rotor System Subcommittee, Design Engineering Division, American Society of Mechanical Engineers, 1972.

2. Hagg, A. C. and Sankey, G. O., "The Containment of Disk Burst Fragments by Cylindrical Shells," ASME Journal of Engineering for Power, April 1974, pp. 114-123.

3. ASTM-A-609-91, Standard Specification for Longitudinal Beam Ultrasonic Inspection of Carbon and Low-alloy Steel Castings.

4. ASTM-E-165-95, Practice for Liquid Penetrant Inspection Method.

5. ANSI/ANS-5.1-1994, "Decay Heat Power in Light Water Reactors."

6. ANSI/ANS-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

7. ANSI N278.1-1975, Self-Operated and Power-Operated Safety-Relief Valves Functional Specification Standard.

8. QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.

9. ANSI B16.34-1996, Valves - Flanged and Buttwelding End.

10. WCAP-15994-P (Proprietary) Revision 1, and WCAP-15994-NP (Non-Proprietary) Revision 1, "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel", March 2003.

Changes to WCAP-15994:

The changes shown in the attached marked-up pages will be incorporated into Revision 1 of WCAP-15994.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

PRA Revision:

None

TABLE OF CONTENTS

5.1.5 Flywheel Enclosure Welds

- Uranium insert normal operation inner radius hoop stress

$$\sigma_{\theta\theta}(R_{OI})_U = \frac{0.688\frac{lb}{in^3}\left(188.5\frac{rad}{sec}\right)^2}{4\times386.4\frac{in}{sec^2}}\left[(3+\nu)(14.50in)^2+(1-\nu)(8.00in)^2\right]=11685psi$$

- Uranium insert design condition inner radius hoop stress

$$\sigma_{\theta\theta}(R_{OI})_U = \frac{0.688\frac{lb}{in^3}\left(1.25\times188.5\frac{rad}{sec}\right)^2}{4\times386.4\frac{in}{sec^2}}\left[(3+\nu)(14.50in)^2+(1-\nu)(8.00in)^2\right]=18255psi$$

The uranium insert is Depleted Uranium Alloy U-2Mo, and the yield stress for this material is 55,000 psi. Per the design specification, the $1/3S_y$ is 18,330 psi and $2/3S_y$ is 36,670 psi. For normal operation at a rotational speed of 1800 rpm, the maximum primary hoop stress is 11,685 psi, which is less than 18,330 psi. Additionally, for a design rotational speed of 2250 rpm, the maximum primary hoop stress is 18,255 psi, which is less than 36,670 psi. Since the rotational stresses for the uranium flywheel are less than the prescribed allowable stresses, the requirements of the design specification are satisfied.

5.1.4 Concentric Ring Elastic Hoop Stresses

In Table 5-2, the hoop stresses at the inner and outer diameter of each of the concentric rings in the model are presented for assembly conditions, 1800 rpm of rotation, operating pressure, and a uniform temperature of 165°F, as well as for assembly conditions and 125-percent overspeed at 70°F. From Table 5-2, it is noted that the hoop stresses of 18.71 ksi and 21.34 ksi in the jacket are less than the yield stresses for Alloy 690, which are 32.5 ksi at 165°F and 35 ksi at 70°F.

INSERT A →

Insert A

5.1.5 Flywheel Enclosure Welds

Since the flywheel enclosure is not considered to be a "reactor coolant pressure boundary", the stress analyses of the enclosure welds has been deferred until the detailed design phase of the AP1000 reactor coolant pump. However, the AP600 and AP1000 flywheel enclosure designs are very similar and the radial thickness of the two enclosures is approximately the same. The stresses calculated in the AP1000 flywheel enclosure shell components at both normal operating and design speeds are similar to those in the AP600 flywheel enclosure. The maximum radial displacement of the AP1000 flywheel enclosure welds is also very similar to that for the AP600 flywheel enclosure welds. The AP600 flywheel enclosure welds have been shown to meet the ASME Code limits during operation at both normal operating and design speeds (Reference 2). Therefore, because of the similarities in enclosure configuration, operating conditions, enclosure shell stresses, radial thickness, and maximum weld area radial displacement, it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits during operation at both normal and design speeds.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.019 (Response Revision 1)

Question:

Section 9.1.3.5, "Safety Evaluation," states that the SFP is designed such that a water level is maintained above the spent fuel assemblies for at least 3 days following a loss of the SFP cooling system, using only safety-related makeup. In the AP600 design, under similar circumstances, water is maintained above the spent fuel assemblies for 7 days (which the NRC staff found acceptable). Please discuss the rationale for maintaining water above the spent fuel assemblies for only 3 days following a loss of the SFP cooling system, and provide the technical justification describing the basis for acceptability of the 3-day duration.

Follow-On Comment:

Clarify whether there is a 3-day or 7-day supply of makeup water for the spent fuel pool.

Westinghouse Response (Revision 1):

Following a loss of spent fuel pool cooling, the water level in the pool is maintained above the fuel assemblies for a minimum of ~~3~~ **7** days utilizng only ~~safety-related~~**on-site** makeup water. **This is consistent with the position that the site be capable of sustaining all design basis events with onsite equipment and supplies for the long term (7 days).** ~~This~~ **It** is **also** consistent with **the coping periods for** both the other AP1000 **safety systems**, and the AP600 passive safety systems~~. design bases. The initial capacities of the class 1E batteries, the passive containment cooling water storage tank, and the control room habitabillity system air storage tanks are sufficient for 72 hours of operation. Consistent with the AP600 treatment of coping periods, the equipment and water sources to extend operation of AP1000 passive safety systems from 72 hours to at least 7 days is available on site. In the case of the spent fuel pool, sufficient water from the seismically qualified passive containment cooling auxiliary water storage tank can be transferred to the spent fuel pool to maintain the water level above the top of the fuel assemblies for a minimum of 7 days, if required. In addition, the non-seismic fire water tank can be used as a source of makeup water for the spent fuel pool.~~

Design Control Document (DCD) Revision:

~~None~~

From DCD Revision 3, page 9.1-20:

9.1.3.5 Safety Evaluation

The only spent fuel pool cooling system safety-related functions are containment isolation and emergency makeup connections to the spent fuel pool. Containment isolation evaluation is described in subsection 6.2.3. The following provides the evaluation of the design of the spent fuel pool as well as the spent fuel pool cooling system:

- The spent fuel pool is designed such that a water level is maintained above the spent fuel assemblies for at least ~~3~~7 days following a loss of the spent fuel pool cooling system, using only ~~safety-related~~**on-site** makeup **water** (see Table 9.1-4). The minimum water level to achieve sufficient cooling is the sub-cooled, collapsed level (without vapor voids) required to cover the top of the fuel assemblies.

- The maximum heat load is assumed to be the heat load for a full core off load immediately following a refueling in which 44% of the fuel assemblies were replaced.

- Safety-related makeup water can be supplied to the fuel pool from the fuel transfer canal, cask washdown pit, and passive containment cooling water storage tank.

- The spent fuel pool cooling system includes safety-related connections from the passive containment cooling system water storage tank in the passive containment cooling system to establish safety-related makeup to the spent fuel pool following a design basis event including a seismic event.

- ~~The spent fuel pool is designed such that a water level is maintained above the spent fuel assemblies for at least 7 days following a loss of the spent fuel pool cooling system, using only on-site makeup water (see Table 9.1-4).~~ In addition to the safety-related water sources, makeup water is also obtained from the passive containment cooling system ancillary water storage tank. ~~After the first three days of the event,~~ W~~w~~ater from this tank can be pumped by the passive containment cooling system recirculation pumps either to the passive containment cooling water storage tank (and then gravity fed to the spent fuel pool), or directly to the spent fuel pool.

Radiation shielding normally provided by the water above the fuel is not required when normal spent fuel pool cooling is not available. Personnel are not permitted in the area when the level in the pool is below the minimum level.

The acceptability of the design of the spent fuel pool cooling system is based on specific General Design Criteria (GDCs) and Regulatory Guides as described in Sections 3.1 and 1.9.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.036 (Response Revision 1)

Original Question:

Section 5.2.2.1 states that a relief valve in the residual heat removal system (RNS) provides low-temperature overpressure protection (LTOP) for the RCS, and that the valve is sized to prevent overpressure based on the following design basis events with a water solid pressurizer:

(1) the limiting mass input event of the makeup/letdown flow mismatch, and
(2) the limiting heat input event of inadvertent start of a reactor coolant pump (RCP).

Provide the safety analyses of both the limiting mass-input and heat-input overpressure events to support the adequacy of the RNS relief valve relieving capacity and set pressure specified in Table 5.4-17 for the LTOP. The description should include:

A. The applicable RCS pressure-temperature limits (LCO 3.4.3) with corresponding neutron fluence values of the reactor vessel, or the effective full power years.

B. The analysis methodology and assumptions, including consideration of limiting single failure assumption, the instrumentation uncertainties of pressure and temperature measurements, the relief valve set pressure and accumulation, the dynamic head effect of the reactor coolant flow, and the static head between the pressure tap and the limiting vessel locations, and pressure overshoot.

C. The analysis results.

D. The determination of the LTOP enable temperature of 275°F (Technical Specifications LCO 3.4.15).

Westinghouse Revision 0 Response:

The normal residual heat removal system (RNS) relief valve mitigates the low temperature overpressure transients and is sized to prevent the RCS pressure from exceeding the lower of either the applicable pressure-temperature (P/T) limit or 110% of the RNS system design pressure. The limiting mass and energy input transients assumed for the sizing analysis are as follows:

- Mass Input: Injection of water into the RCS from the operation of both makeup pumps due to makeup/letdown flow mismatch. The maximum flow mismatch is 177 gpm. The makeup flow is limited by the cavitating venturi located in the discharge header of the chemical and volume control system makeup pumps.

- Energy Input: During an RCS cooldown, the reactor coolant pumps are tripped at an RCS temperature of approximately 160 F. Below this temperature, the RNS continues to cool down the RCS, while the steam generators may remain at or near 160 F. It can be postulated that a 50 F differential temperature can be developed between the RCS and the steam generators under this condition. Subsequent restart of one reactor coolant pump under these conditions results in the limiting energy input cold overpressure transient. This transient is postulated to occur over a range of reactor coolant temperatures between 100 F and 200 F because an administrative requirement has been imposed in the Technical Specifications that does not allow a reactor coolant pump to be started while the RCS is water solid and the RCS temperature is above 200 F.

A. The nominal steady-state P/T limits applicable up to 54 effective full power years (EFPY) are given in DCD Figures 5.3.2 and 5.3.3. The lowest Appendix G limit from these curves is 1023 psig. The RNS system design pressure is 900 psig, and therefore the system pressure limit is 990 psig. Therefore, the lowest of the two pressure limits (990 psig) is used as the limit in the sizing of the RNS relief valve.

B. & C. The energy input transient is the limiting event for an RCS temperature above 100 F. Below 100 F, the mass input transient is more limiting. The energy input transient is analyzed using a specialized version of the LOFTRAN computer code (Reference 1), which has the capability to model the RNS relief valve. The peak pressure in the RNS system is calculated using the methodology as described in Reference 2 except that the RNS relief valve instead of the pressurizer PORV is used to mitigate the energy input transient.

Based on the energy input transient, the minimum RNS relief valve capacity of 750 gpm has been calculated at an RCS pressure equivalent to the valve setpoint of 636 psig plus 10% accumulation (700 psig). With this setpoint and capacity, the relief valve mitigates the limiting LTOP transient while maintaining the RCS pressure less than 110% of RNS design pressure. Since the relief valve is located on the RNS pump suction line, the set pressure must account for the RNS pump head to maintain the RNS discharge piping below the system design pressure. Pressure losses in the flow path and the static pressure difference between the RNS suction piping and the relief valve are also considered in establishing the relief valve set pressure. The peak pressure at the discharge of the RNS pump for the energy input transient is no higher than 979 psig. The peak pressure at the inlet to the RNS relief valve is 779 psig.

The minimum required capacity of the RNS relief valve based on the energy input transient is 750 gpm. Since the maximum flow rate for the mass input transient is 177 gpm, the RNS relief valve will be adequate to mitigate the mass input transient without overpressurizing the RNS system. The peak pressure at the inlet to the RNS relief valve will be no higher than the RNS relief valve full open pressure of 700 psig. The peak pressure at the discharge of the RNS pump will be no higher than 900 psig.

Single active failure is not considered for passive valves such as the RNS self-actuated spring relief valve. Therefore, the analysis does not consider a single failure of this valve. Also, no single active failure can occur in the RNS that could prevent the RNS suction relief valve from performing its function.

The 10% setpoint accumulation includes a 3% setpoint uncertainty. No other uncertainties are explicitly modeled in the analysis.

D. The LTOP enable temperature is based on utilizing the pressurizer safety valves for RCS overpressure protection when the RCS temperature is above 275 F (Technical Specification LCO 3.4.15). Once the RCS temperature reaches 275 F the RCS pressure can exceed the pressurizer safety valve set point pressure (2500 psig) and still be in the acceptable operating range according to the pressure/temperature curves (DCD Figures 5.3-2 and 5.3-3). The RCS pressure transients described in DCD section 15.2.3 confirm that the pressurizer safety valves are adequately sized to provide RCS overpressure protection.

DCD Table 5.4-17 will be revised to reflect the RNS relief valve design parameters given above.

Reference 1: WCAP-7907-PA, "LOFTRAN Code Description," April 1984.

Reference 2: WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1998.

Design Control Document (DCD) Revision:

From DCD page 5.4-40: (These changes were incorporated into DCD Revision 3)

5.4.7.1.2.5 Low Temperature Overpressure Protection

The normal residual heat removal system provides a low temperature overpressure protection function for the reactor coolant system during refueling, startup, and shutdown operations. The system is designed to limit the reactor coolant system pressure **to the lower of either** the limits specified in 10 CFR 50, Appendix G, **or 110% of the normal residual heat removal system design pressure.**

From DCD page 5.4-61, Section 5.4.9.3: (These changes were incorporated into DCD Revision 3)

The relief valve on the normal residual heat removal system has an accumulation of 10 percent of the set pressure. The set pressure is the lower of the pressure based on the design pressure of the residual heat removal system and the pressure based on the reactor vessel low temperature pressure limit. The pressure limit determined based on the design pressure includes the effect of the pressure rise across the pump.

The set pressure in Table 5.4-17 is based on the ~~reactor vessel low-temperature pressure limit~~**design pressure of the residual heat removal system**. The lowest permissible set pressure is based on the required net positive suction head for the reactor coolant pump.

From DCD page 5.4-93: (These changes were incorporated into DCD Revision 3)

Table 5.4-17

PRESSURIZER SAFETY VALVES - DESIGN PARAMETERS

Number	2
Minimum required relieving capacity per valve (lb/hr)	750,000 at 3% accumulation
Set pressure (psig)	2485 ±25 psi
Design temperature (°F)	680
Fluid	Saturated steam
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	500
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100

Residual Heat Removal Relief Valve - Design Parameters

Number	1
Nominal relieving capacity per valve, ASME flowrate (gpm)	~~650~~**750**
Nominal set pressure (psig)	~~818~~**636***
Full-open pressure, with accumulation (psig)	~~900~~**700***
Design temperature (°F)	400
Fluid	Reactor coolant
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	200
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100

* See text (5.4.9.3) for discussion of set pressure

PRA Revision:

None

NRC Additional Questions:

(A) Considering the pressure difference between the RCS and the RNS, what is the equivalent RCS pressure at the RNS pressure of 990 psig?

(B) DCD Figures 5.3-2 and 5.3-3 indicates that the P/T limits in these figures are without margins for instrumentation errors. What would the lowest RCS pressure limit be with the instrumentation errors accounted for in the P/T limit curves? Is it higher than the equivalent RCS pressure of 110% RNS design pressure?

(C) Explain why the peak pressures for the energy input transient are 979 psig and 779 psig, respectively, at the discharge of the RNS pump and the inlet to the RNS valve. What is the peak pressure in the RCS?

Westinghouse Response to Additional Questions:

(A) The maximum pressure in the RCS (with one reactor coolant pump operating) is approximately 850 psig when the maximum RNS pressure is 990 psig. The maximum RCS pressure occurs at the discharge of the operating reactor coolant pump.

(B) Assuming a 60 psi RCS pressure instrument error, the lowest RCS pressure limit would be 963 psig (1023 psig which is the lowest Appendix G limit from DCD Figures 5.3.2 and 5.3.3 minus the 60 psi instrument error). This pressure is higher than the equivalent maximum RCS pressure (850 psig) when the RNS pressure is 110% of system design pressure (990 psig).

Note that low temperature overpressure protection is provided in the AP1000 RNS by a spring loaded relief valve which is self-actuated by direct fluid pressure. Therefore, actuation of the relief valve is not dependent on pressure measurements in the RCS or RNS.

(C) The RNS relief valve set pressure is established based on the applicable limits of either RNS design pressure or the 10 CFR Appendix G limits. For the AP1000, the lowest Appendix G limit is 1023 psig (DCD Figures 5.3.2 and 5.3.3) and the RNS system design pressure is 900 psig. Since the RNS system design pressure is more limiting, the RNS relief valve set pressure is established such that the valve full-open pressure plus the

pressure differential between the valve inlet and the maximum pressure point in the RNS system is no higher than 110% of the RHR system design pressure. Since the RNS relief valve is located on the RNS pump suction piping, the set pressure must account for the RNS pump head and elevation differences between the relief valve and the pump to maintain the RNS pump discharge piping below 110% of RNS system design pressure.

The energy input transient was analyzed for the AP1000 using the methodology of WCAP-14040-NP-A (Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves) except that the RNS relief valve instead of the pressurizer PORV is used to mitigate the energy input transient. This analysis showed that by establishing a RNS relief valve nominal set pressure of 636 psig with a capacity of 750 gpm, the maximum pressure that results at the inlet to the relief valve is 779 psig. The resulting maximum pressure in the RNS at the discharge of the RNS pump is 979 psig, which is below the limit of 110% of system design pressure (990 psig).

The peak pressure in the RCS corresponding to the RNS conditions above is approximately 840 psig. This pressure would occur at the discharge of the one reactor coolant pump assumed to be operating.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.154 (Response Revision 2)

Question:

Section A.4.4 presents results of a sensitivity study using the Top Offtake Orientation model. In general, the onset for entrainment for a top offtake configuration is:

$$Fr_g = \frac{U_g}{\sqrt{\frac{gd\Delta\rho}{\rho_g}}} = C_1\left(\frac{h_b}{d}\right)^{C_2} \qquad (1)$$

where d is the diameter of the branch line.

In the sensitivity study, the values of C_1 and C_2 were varied from their reference values of C_1=0.355 and C_2=2.5. The results were intended to show that the AP1000 performance for the Inadvertent ADS case has little sensitivity to the Top Offtake Orientation correlation.

(a) Since hot leg entrainment is sensitive to steam velocities in the hot leg, show why the Inadvertent ADS case is sufficient to characterize or bound AP1000 performance for other accident scenarios such as a small cold leg break or a direct vessel injection (DVI) line break.

(b) When either the coefficient C_1 is increased or the exponent C_2 is decreased in the sensitivity study, the minimum in-vessel inventory increases and the start of in-containment refueling water storage tank (IRWST) injection is delayed compared to the Reference case. However, when C_1 is decreased, the vessel inventory minimum again increases and the IRWST injection is delayed. Please explain why both increasing and decreasing the onset of entrainment results in this sensitivity.

(c) Provide justification that the variation of the Top Offtake Orientation model in Section A.4.4 is sufficient to account for inaccuracies in the model when compared to experimental data. Figure 2 shows the variations considered in the sensitivity study. When data from ATLATS tests are compared to the correlation and its variations in the sensitivity study, it can be seen that the branch line quality is grossly over-estimated for the two cases in which the offtake quality was reduced compared to the Reference. For example, at Froude numbers less than one the data suggests very low branch line quality while the correlation and its variations predict very high quality (x >0.75).

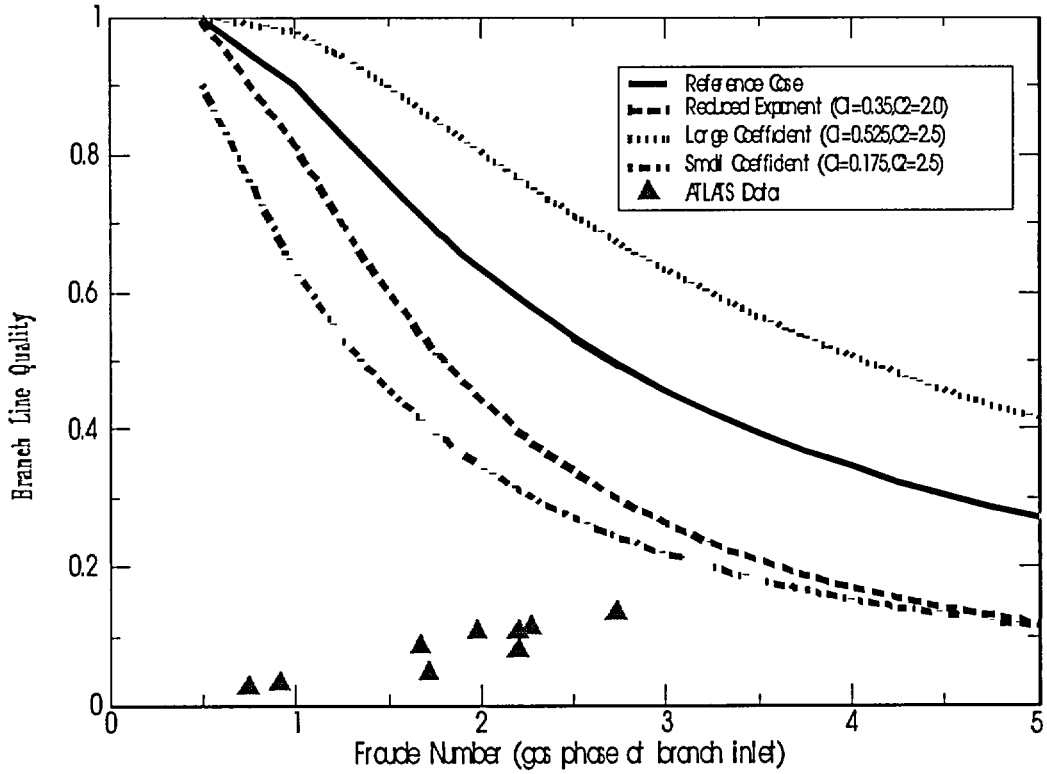


Figure 2. Variation in Branch Line Quality of Entrainment Onset Correlations in Westinghouse Sensitivity Study

Westinghouse Response (Revision 1):

a) A comprehensive set of figures is provided in the response to RAI 440.163 for both the DEDVI break case and the Inadvertent ADS scenario. The flow regime maps for both hot legs are shown in RAI 440.163 for both of the cases. The hot leg behavior of the DEDVI case is such that during the interval from the time at which all the ADS-4 flow paths have opened to the beginning of IRWST injection, the flow regime is almost always horizontal stratified flow. In contrast, there are instances of countercurrent flow in the Inadvertent ADS case during this interval, due at least in part to lower hot leg velocities. Therefore, there are greater flow regime variations predicted for the Inadvertent ADS scenario during the most important portion of the depressurization transient. In order to study the impacts of perturbations over a wider hot leg flow regime range, the judgment is that the Inadvertent ADS scenario is an appropriate case to characterize the sensitivity of AP1000 performance

to parameter variations in WCOBRA/TRAC. Small RCS loop break cases would likely be little different from the Inadvertent ADS case in behavior.

b) The attached Figure 440.154-1 compares the downcomer pressure during the ADS-4 IRWST initiation phase for the increased entrainment onset correlation coefficient (solid line) and the decreased entrainment onset correlation coefficient (dashed line). The time scale is the same as in Appendix A.4 of WCAP-15833. Prior to 65 seconds, only one ADS-4 flow path is open. At 65 seconds, the pressure in the small coefficient case is about 3 psi higher than in the large coefficient case. This result is consistent with Figure 154-2 in that the higher branch line quality associated with the large coefficient might be expected to produce a more effective depressurization of the AP1000 than that observed in the small coefficient case, with its lower quality. However, once the additional two ADS-4 flow paths become available at 65 seconds, the pressures converge. This indicates that the AP1000 design has adequate depressurization capability that, even assuming a single failure, the pressure result during ADS-4 operation is not sensitive to the variation in the entrainment onset correlation coefficient.

Figure 440.154-2 compares the hot leg levels in the loop that contains at least one open ADS-4 flow path throughout the ADS-4 IRWST initiation phase for the increased entrainment onset correlation coefficient (solid line) and the decreased entrainment onset correlation coefficient (dashed line) cases. Prior to 65 seconds a difference is established with the reduced coefficient case exhibiting a lower level, which is equivalent to a larger distance "h" between the top of the hot leg and the liquid surface. However, once the additional two ADS-4 flow paths are available at 65 seconds, the collapsed levels become very similar, again indicating a lack of sensitivity to the entrainment onset correlation. The minimum mass inventory results of these cases and the base case are all within 1000 lbm of one another out of a total minimum inventory of 71000 lbm.

c) As noted in the response to RAI440.151, it is reasonable to expect the form of the correlation in WCOBRA/TRAC to remain valid for the AP1000 ADS-4 configuration, with only variation in the coefficient (C1) or exponent (C2). To investigate behavior, sensitivity calculations of entrainment from the hot leg to the ADS-4 offtake in AP1000 were performed with the WCOBRA/TRAC-AP models where the coefficient (C1) and exponent (C2) associated with the entrainment inception correlation were varied as described in Appendix A.4 of WCAP-15833. These calculations indicated that the horizontal flow regime for AP1000 during the ADS-IRWST transition phase of a SBLOCA is primarily stratified in the hot legs upstream of the ADS-4 off-takes. Hence, the correlations are applicable. Also, there is little sensitivity to variations in the coefficient (C1) or in the exponent (C2) associated with the entrainment inception correlation. At the AP1000 Froude number of ≥ 2, a variation of $\pm$ 50% in the branch line quality value is indicated in the figure above for the sensitivity study investigation performed with WCOBRA/TRAC-AP. This is judged to be an adequate

range for the studies performed varying parameters in the hot leg entrainment correlations of WCOBRA/TRAC-AP.

With respect to addressing the data associated with the ATLATS separate effects test facility, Westinghouse does not have detailed information regarding these tests (i.e. facility layout, boundary conditions, test procedures, scaling analysis, etc.) and therefore these tests have not been analyzed by Westinghouse. A possible explanation of the different behavior displayed in ATLATS relative to the stratified flow type entrainment behavior expected in AP1000 and seen in other test facilities is that the ATLATS test facility is producing a different flow regime that may be attributed to its non-prototypic and/or incomplete simulation of the actual AP1000 configuration.

NRC Additional Comments

The response to part (c) is not sufficient, since the response to RAI 440.151 was not considered sufficient. The response to part (c) of this RAI implies that because the code predicted the flow pattern to be horizontal-stratified, a correlation of the form of Equation 2-24 of WCAP-15833 must be applicable.

It is not clear why the 50% ranging on branch line quality performed in the sensitivity study is adequate. Early in the DEDVI and Inadvertent ADS transients discussed in the response to RAI 440.163, the mass flow rate into the hot legs are nominally between 20 and approximately 70 lbm/sec (9 and 32 kg/sec). At 22 psia (0.152 MPa), the Froude number at the ADS-4 branch line will vary between approximately 1.5 and 5.5. In the Inadvertent ADS case, it is generally less that 3.0. The response does not provide sufficient information on why the ranging performed in the sensitivity study is adequate from these low Froude numbers. A convincing argument that hot leg phase separation is not a dominant process in AP1000 may be made if the ranging were performed such that branch line flow quality at low Froude numbers was more strongly affected.

The response to part (c) also discusses the data from the ATLATS facility. These data were obtained as part of an experimental investigation performed at Oregon State University. The vessel, steam generator inlet plenum, and hot leg of the ATLATS test facility itself is scaled 1:1 with the APEX facility, and approximate flow phenomena that occurred in the APEX integral tests. If ATLATS is non-prototypical of AP1000, then additional information on why the APEX tests remain adequately scaled to AP1000 should be provided, since APEX would have the same problem. A description of the ATLATS facility, data, and test procedures is available upon request. The NRC requests that the Figures provided in this RAI response be included in WCAP-15833.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Westinghouse Response Revision 2

Please see the Westinghouse Revision 1 Response to RAI 440.151 that was transmitted in Westinghouse letter DCP/NRC1555 dated March 17, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

Westinghouse will revise WCAP-15833 to include an appendix that contains the Westinghouse responses to NRC RAI related to WCAP-15833.

Figure 440.154-1 Downcomer Pressure Comparison, Hot Leg Entrainment Onset Study

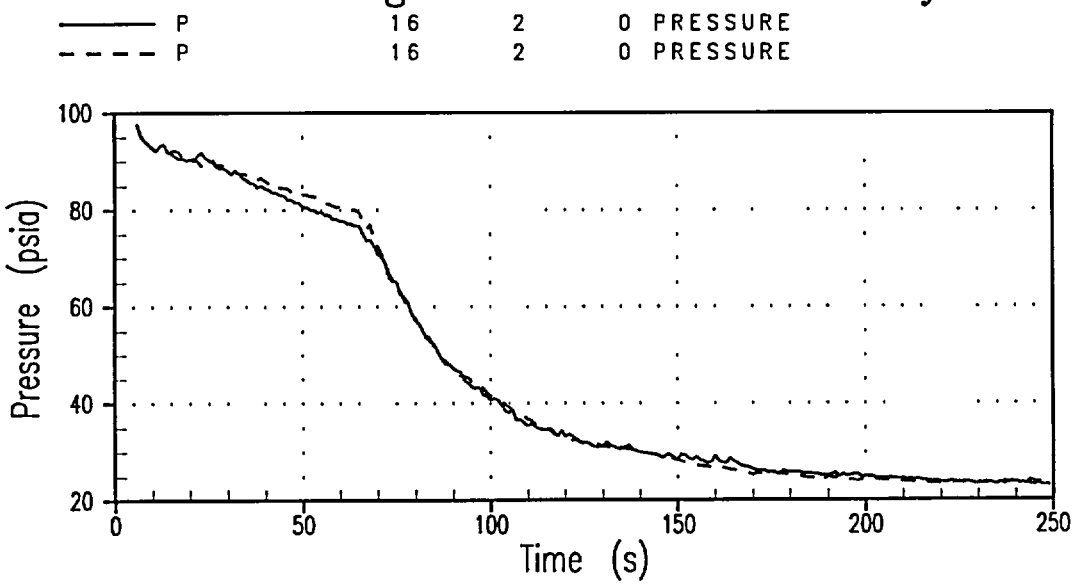
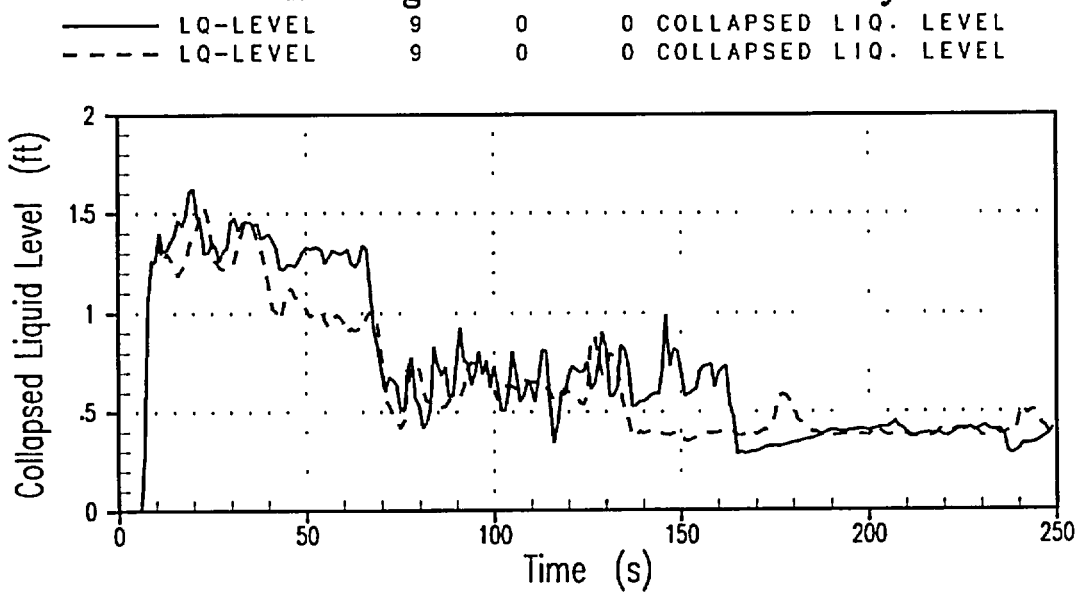


Figure 440.154-2 2*ADS-4 Hot Leg Collapsed Liquid Level, Hot Leg Entrainment Onset Study



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.157 (Response Revision 1)

Question:

As described in Section 2.2.1.4 for horizontal stratified flow, the correlation by Tatterson, et al. [1] is used to determine the size of entrained droplets. In the original reference, it was recommended that the "volume median diameter" be approximated as,

$$\frac{D_d}{D_t}\left(\frac{\rho_g U_g^2 f_s D_t}{2\sigma}\right)^2 = 0.016 \qquad (1)$$

where D_d is the volume median diameter, D_t is the channel hydraulic diameter, and f_s is the friction factor for a smooth interface.

(a) The coefficient in Equation (2-20) of WCAP-15833 does not appear to be correct. Please demonstrate that Equation (2-20) is equivalent to the expression recommended by Tatterson, et al. [1].

(b) Provide justification for using the hydraulic diameter of the gap above the mixture elevation as the characteristic length instead of the channel hydraulic diameter as recommended by Tatterson, et al.

(c) For the velocities and conditions in the hot legs of AP1000 and the APEX tests for which Equation (2-20) is applied, provide justification that the predicted drop diameters are in reasonable agreement with experimental data. Note that Tatterson, et al., did in fact list drop diameter for horizontal flows for an air-water system and the flow rates.

Reference

[1] Tatterson, D. F., Dallman, J. C., and Hanratty, T. J., ADrop Sizes in Annular Gas-Liquid Flows,@ 1977, AIChE Journal, Vol. 23, No. 1, pp. 68-76.

Westinghouse Revision 0 Response:

In section 2.2.1.4 of WCAP-15833 it was incorrectly stated that the size of the entrained droplets for the horizontal stratified flow is determined by Tatterson's model. The correlation used in WCT (Eq. 2-20 of WCAP-15833) to determine the characteristic size of the entrained droplet within the hot leg is not the same or equivalent to Tatterson's correlation.

Nevertheless the following discussion shows that the model used to determine the droplet size for the horizontal stratified flow regime is similar to the Tatterson's model. Moreover, for the expected conditions in the AP1000, the droplet size under question is a secondary effect such that any bias does not affect the system response significantly.

Within the code the droplet size is used to determine the interfacial area concentration source term based on Equation 4-125 of the CQD:

$$A_{i,E}''' = \frac{6S_E}{\rho_l D_{d,E}} - \frac{6S_{DE}}{\rho_l D_d}$$

After the solution of the interfacial area transport equation (Equation 3-72 CQD), the local droplet size is back calculated from the local interfacial area concentration as follows:

$$D_d = \frac{6\alpha_e}{A_i'''}$$

As a result the droplet size $D_{d,E}$ affects the local characteristic droplet size D_d depending on how significant is the source of entrainment in the horizonthally stratified hot leg relative to the net entrainment flow from the upper plenum. At the inlet boundary of the hot leg, the entrained field is characterized by a droplet size, which is calculated based on the entrainment process occuring in the upper plenum. Inside the hot leg the local droplet size can vary because of the entrainment process occuring within the horizontal stratified section of the hot leg. The interfacial area transport equation attempts to represent both population of droplets with a characteristic droplet size calculated from the solution of the interfacial area transport equation. When the entrainment flow rate at the boundary of the cell is dominant over the local source of entrainment and de-entrainment the effect of the droplet size $D_{d,E}$ is small.

Figure 1 shows the entrainment flow rate at the inlet of one of the hot legs during a time window after the opening of ADS 4. Figure 2 shows both the sources of entrainment and de-entrainment in Channel 24. Channel. 24 is the first hot leg channel and is located upstream of Channel. 25 which is connected to the ADS 4 line. The comparison in Figure 3 shows clearly that based on the WCT predictions the entrainment flow from the upper plenum is the dominant component. Predictions show that most of the liquid is in the dispersed form. Figure 4 shows that the continuous liquid field volume fraction (film) is very small through the transient. As a result, the effect of entrainment and de-entrainment in the hot leg is expected to be small, and the characteristic size of the droplet is also entrained within the hot leg is very small.

Measurements of drop size in horizontal two-phase flow are sparse and typically results for a vertical configuration are used. The droplet size predicted by Eq. 2-20 (Dwct) of WCAP-15833 was compared with the original Tatterson (Dtat) and two other recent correlations suggested for annular flow by L. Pan and Hanratty (2002) (D25 and D26). The calculation is provided in the attachment. In the calculation it was assumed that Dg~Dh. The droplet size calculated with Eq.

2-20 of WCAP-15833 is about 30% smaller than the one predicted with Tatterson. On the other hand the two correlations suggested by Pan and Hanratty (2002) give droplet size that are significantly smaller than Tatterson.

In conclusion, for the conditions expected in AP1000, the droplet size predicted with Eq. 2-20 of WCAP-15833 is in reasonable agreement with test data. Moreover any bias has a very small effect on the transient response because the source of entrainment within the hot leg is expected to be small, compared to the entrainment flow from the upper plenum is the dominant component.

Reference:

[2] Lei Pan, T. Hanratty, "Correlation of entrainement for annular flow in horizontal pipes", Int. J. of Multiphase Flow 28 (2002) 385-408.

NRC Additional Comment

Regarding the response supplied in the November 26, 2002, transmittal memo (W Ref.: DCP/NRC1535), revisions must be made to 2.2.1.4 to eliminate the incorrect statement.

Westinghouse Revision 0 Response:

WCAP-15833 will be revised as shown.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

The size of the entrained droplets is determined by **the following equation which is similar to** Tatterson's (Reference 13) model:

$$D_e = 0.0112\left(\frac{D_g\sigma}{0.5 f_1 \rho_v U_v^2}\right)^{1/2} \qquad (2\text{-}20)$$

Also note that WCAP-15833 will be revised to include an Appendix containing the Westinghouse responses to NRC RAI related to WCAP-15833.

ATTACHMENT

Entrainment in Horizontal Stratified Flow
- Droplet Size -

Inputs:

$wg := 50 \frac{lb}{s}$

Fluid Properties :

$p := 550000\ Pa$

$\rho g := 2.91 \cdot \frac{kg}{m^3}$ $\quad \rho f := 912 \cdot \frac{kg}{m^3}$ $\quad \mu g := 0.0000142 \cdot Pa \cdot s$ $\quad \mu f := 0.000176 \cdot Pa \cdot s$

$\sigma := 0.047 \cdot \frac{N}{m}$

Geometry:

$Dh := 0.74 \cdot m$

$\alpha := 0.8$ $\quad Ag := \alpha \cdot \left(\pi \cdot \frac{Dh^2}{4} \right)$ $\quad Dg := Dh$

Calculation:

$ug := \frac{wg}{\rho g \cdot Ag}$ $\quad ug = 22.652 \frac{m}{s}$

$fi := 0.046 \left(\frac{\rho g\ ug \cdot Dg}{\mu g} \right)^{-0.2}$ $\quad fi = 2.268 \times 10^{-3}$

$Dwct := 0.0112 \left(\frac{Dg \cdot \sigma}{0.5 fi \cdot \rho g \cdot ug^2} \right)^{0.5}$ $\quad Dwct = 1.605 \times 10^{-3}\ m$

$Dtat := 0.016 \cdot Dh \cdot \left(\frac{\sigma}{0.5 fi\ \rho g \cdot ug^2 \cdot Dh} \right)^{0.5}$ $\quad Dtat = 2.293 \times 10^{-3}\ m$

$D25 := \left(0.0091 \frac{\sigma\ Dh}{\rho g \cdot ug^2} \right)^{0.5}$ $\quad D25 = 4.604 \times 10^{-4}\ m$

$D26 := \left[0.14 \left(\frac{\sigma}{\rho f \cdot g} \right)^{0.5} \cdot \frac{\sigma}{\rho g \cdot ug^2} \right]^{0.5}$ $\quad D26 = 1.005 \times 10^{-4}\ m$

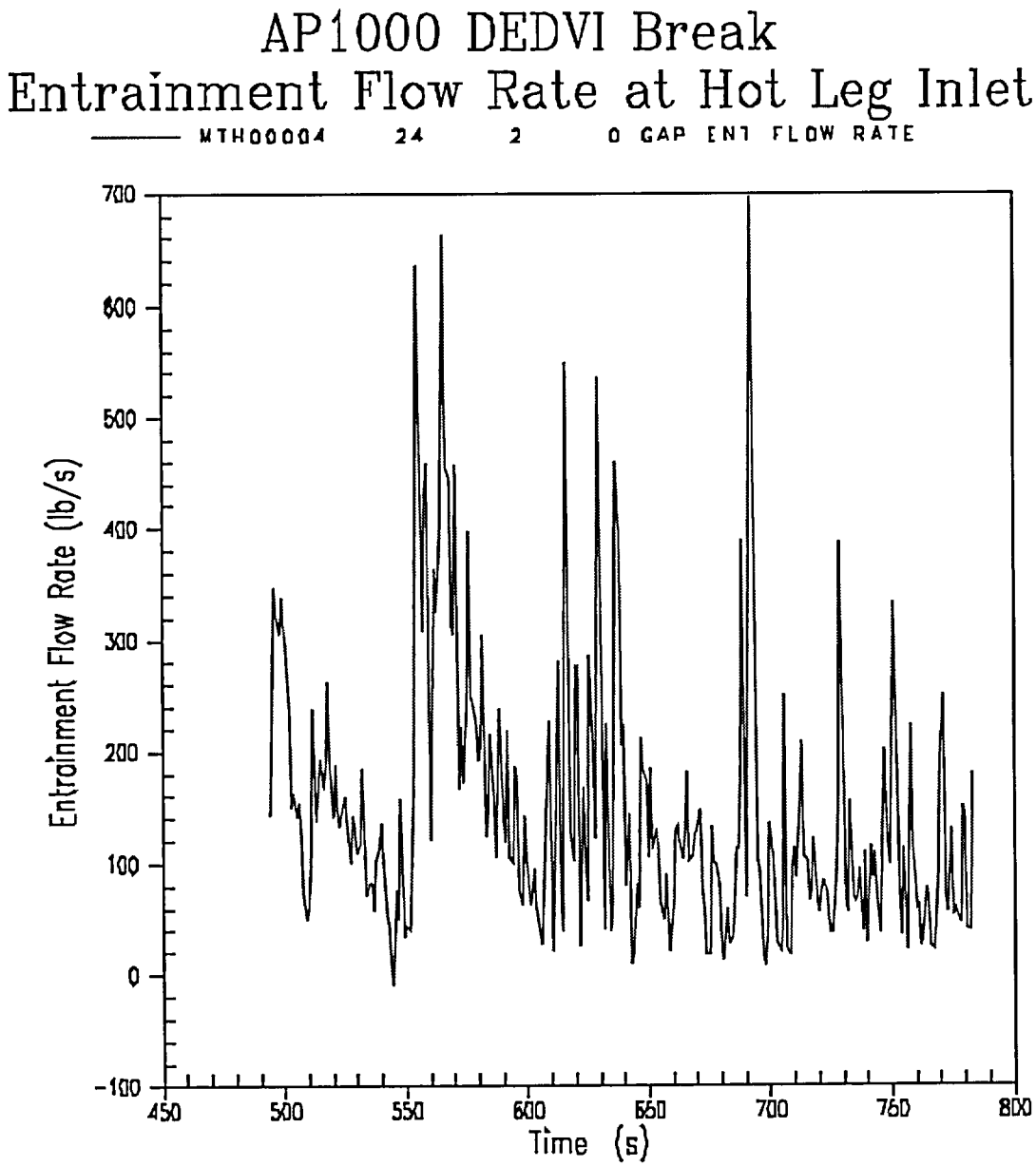

AP1000 DEDVI Break
Entrainment Flow Rate at Hot Leg Inlet
MTH00004 24 2 0 GAP ENT FLOW RATE
Entrainment Flow Rate (lb/s)
700
600
500
400
300
200
100
0
-100
450
500
550
600
650
700
750
800
Time (s)

Figure 1

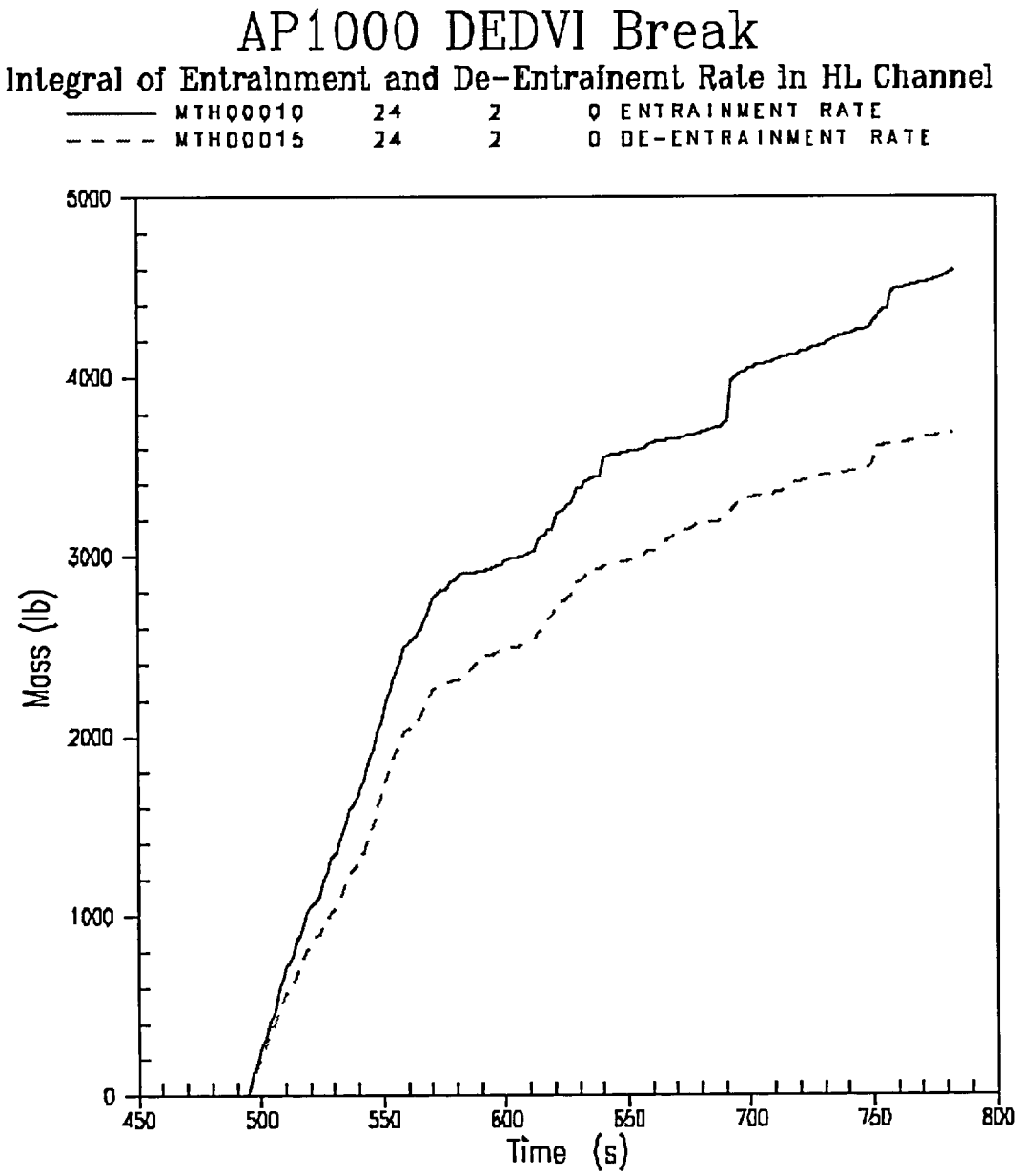

AP1000 DEDVI Break
Integral of Entrainment and De-Entrainemt Rate in HL Channel
MTH00010 24 2 0 ENTRAINMENT RATE
MTH00015 24 2 0 DE-ENTRAINMENT RATE
Mass (lb)
Time (s)
0
1000
2000
3000
4000
5000
450
500
550
600
650
700
750
800

Figure 2

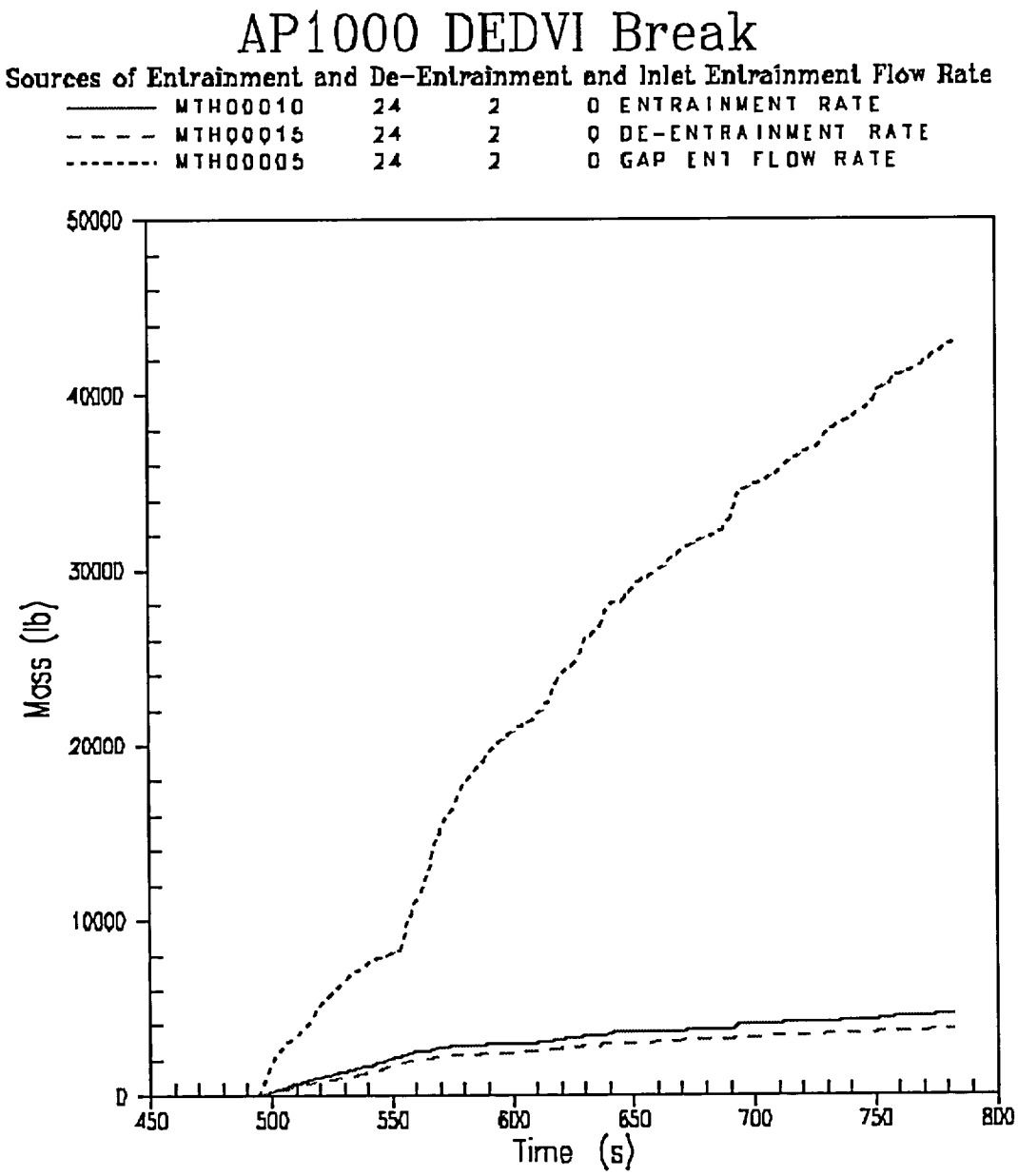

AP1000 DEDVI Break
Sources of Entrainment and De-Entrainment and Inlet Entrainment Flow Rate
MTH00010 24 2 0 ENTRAINMENT RATE
MTH00015 24 2 0 DE-ENTRAINMENT RATE
MTH00005 24 2 0 GAP ENT FLOW RATE
50000
40000
30000
20000
10000
0
Mass (lb)
450
500
550
600
650
700
750
800
Time (s)

Figure 3

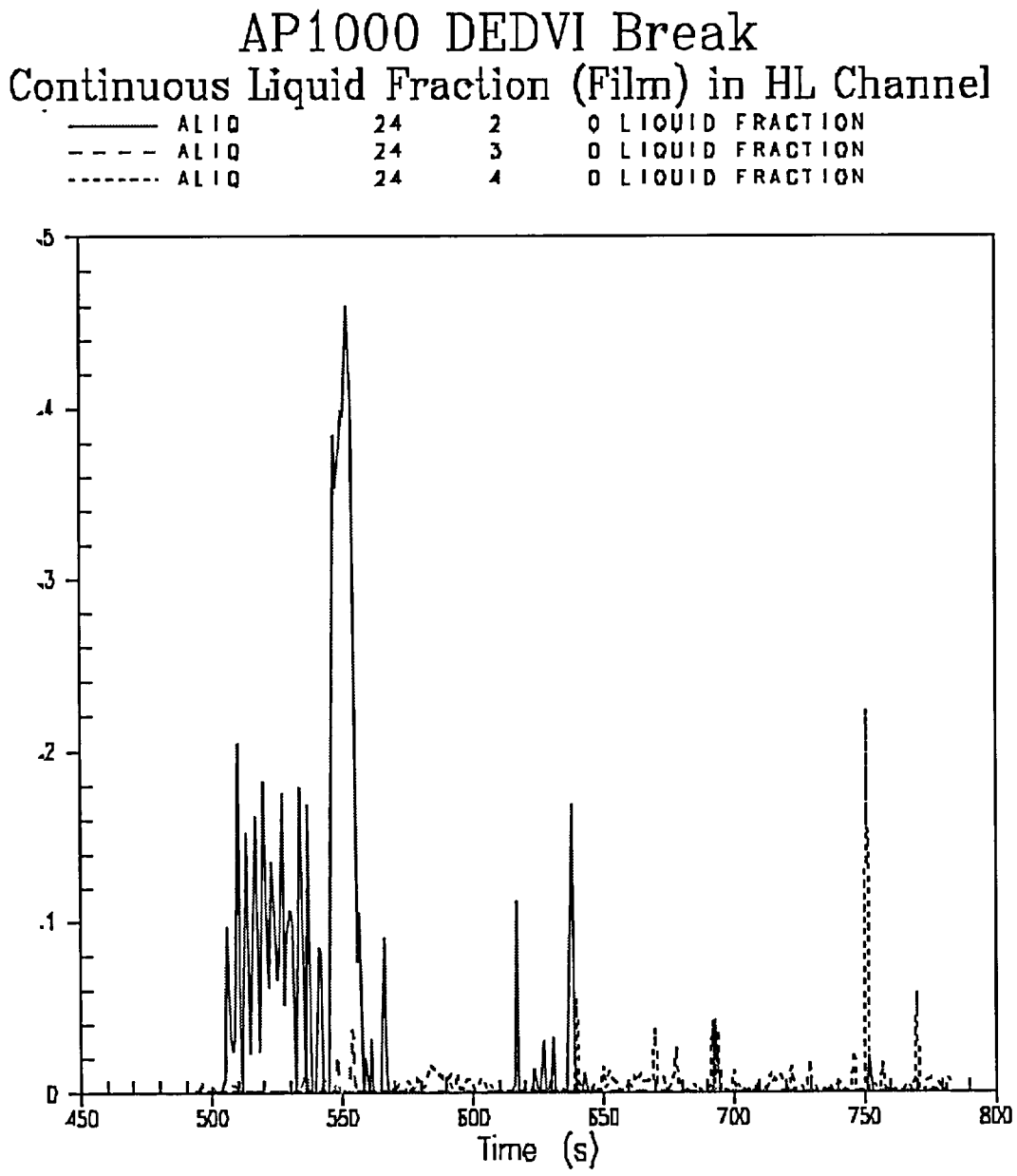

AP1000 DEDVI Break
Continuous Liquid Fraction (Film) in HL Channel
ALIQ 24 2 0 LIQUID FRACTION
ALIQ 24 3 0 LIQUID FRACTION
ALIQ 24 4 0 LIQUID FRACTION
.5
.4
.3
.2
.1
0
450
500
550
600
650
700
750
800
Time (s)

Figure 4

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

AI Number: 440.160 (Response Revision 1)

Question:

WCAP-15833

Please provide Figures 2-9, 2-11, and 2-13 that include a key, or clearly indicate what the individual curves represents. In addition, explain why there are periodic oscillations in the steam flow rate (see dashed curves in Figure 2-13) although the pressure differentials along the channel remain relatively constant (see Figure 2-11).

Westinghouse Response:

Figure 2-9 of WCAP-15833 presents the calculated liquid levels for the individual gaps in the WCOBRA/TRAC nodal network. Gaps connect adjacent channels within a section of a WCOBRA/TRAC VESSEL component model at each vertical cell; Figure 2-6 of WCAP-15833 indicates the gap numbering scheme for the Lim test facility as **G2, G4**, etc. An expanded version of the WCAP-15833 plot with each curve identified is attached as Figure 440.160-1. Figure 2-11 presents the calculated steam pressure of [ ]^{a,c} individual channels in the WCOBRA/TRAC nodal network. An expanded version of this WCAP-15833 plot is provided as Figure 440.160-2, wherein each curve is identified individually. Figure 2-13 presents the calculated steam flow rate [

]^{a,c}

Figure 440.160-3 exhibits periodic oscillations in the steam flow rate [

]^{a,c} the results are very stable at all gap locations, as shown in the attached Figure 440.160-4. The variation in the [

]^{a,c}

NRC Additional Comment:

Regarding the response supplied in the November 1, 2002, transmittal memo (W Ref.: DCP/NRC1529), Figures 2-9, 2-11, and 2-13 in the RAI response must replace the originals in a revision to WCAP-15833.

Westinghouse Revised Response:

The proprietary Figures 2-9, 2-11, and 2-13 in the RAI response will be added to the proprietary version of WCAP-15833. The proprietary Figures will be included in the Appendix that is being added that contains the Westinghouse responses to NRC RAI related to WCAP-15833.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

WCAP-15833 will be revised to include an Appendix containing the Westinghouse responses to NRC RAI related to WCAP-15833.

a,c

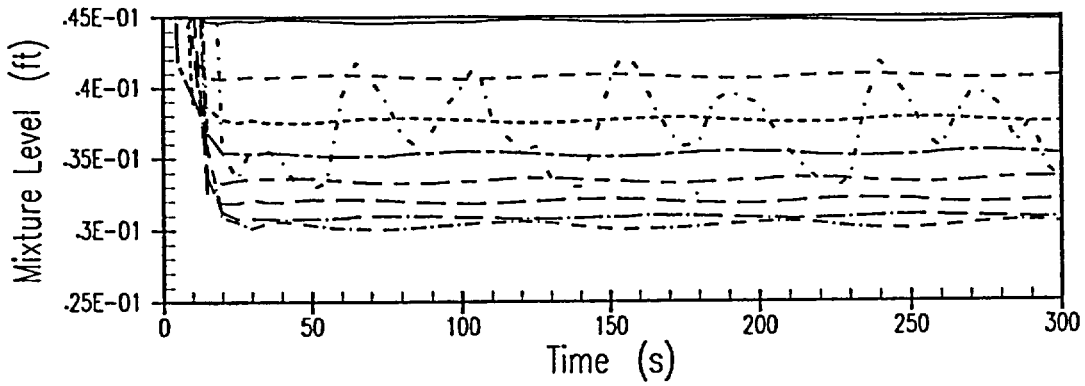


FIGURE 440.160-1: CALCULATED LIQUID LEVEL

a,c

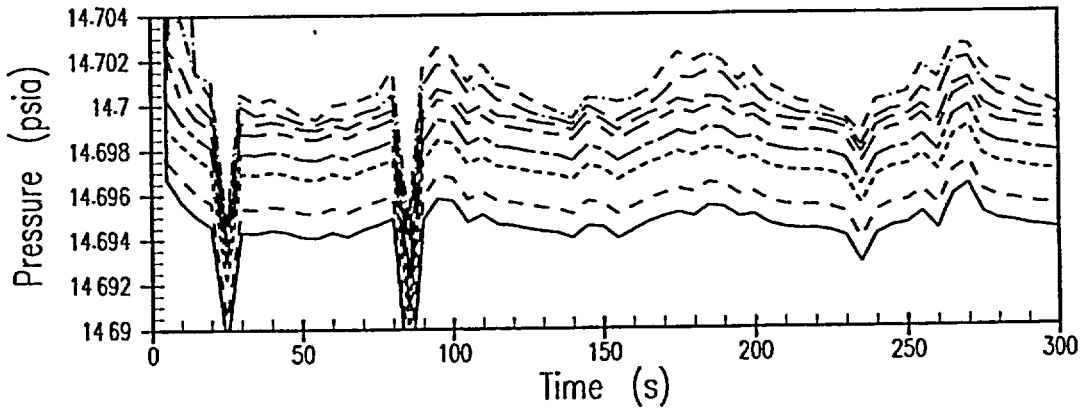


FIGURE 440.160-2: CALCULATED PRESSURE

a,c

FIGURE 440.160-3:

Calculated Steam Flowrate (Run 275)

a,c

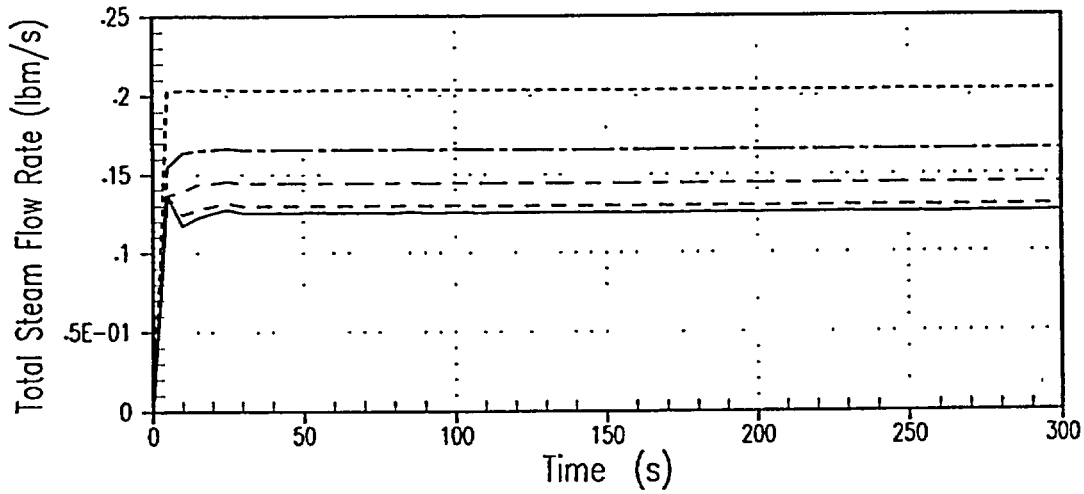


FIGURE 440.160-4: TOTAL GAP STEAM FLOW RATE

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.171 (Response Revision 1)

Question:

Figures A.3-4 and A.3-5 show a prediction of WCOBRA/TRAC upper plenum entrainment versus time for conditions simulating a DVI line break. Please provide the WCOBRA/TRAC collapsed liquid level for this simulation.

Westinghouse Response:

The collapsed liquid level in the upper plenum from the WCOBRA/TRAC DEDVI line break simulation presented in Appendix A.3 is shown in Figure 440.171-1. The level shown is relative to the top elevation of the active fuel length.

NRC Additional Comment:

Regarding the response supplied in the November 1, 2002, transmittal memo (W Ref.: DCP/NRC1529), the figure must be included in a revision to WCAP-15833.

Westinghouse Additional Response:

WCAP-15833 will be revised to include an Appendix containing the Westinghouse responses to NRC RAI related to WCAP-15833. This Appendix will therefore include the Figure.

Design Control Document (DCD) Revision:

None

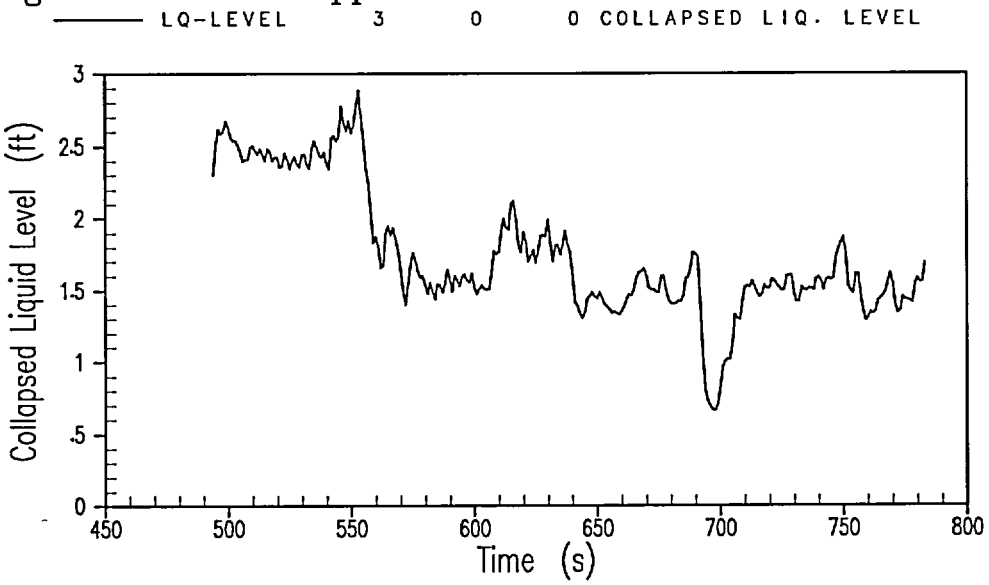
PRA Revision:

None

WCAP Revision:

WCAP-15833 will be revised to include an Appendix containing the Westinghouse responses to NRC RAI related to WCAP-15833.

Figure 440.171–1: Upper Plenum Level, DEDVI Break Case



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.186

Question:

Westinghouse indicated in DCD Section 19E.4.7.1 that the loss of RCS inventory aspects of the inadvertent opening of a pressurizer safety valve and inadvertent actuation of automatic depressurization valves events is covered in Subsection 19E.4.8.2 (or 19E.4.8 as corrected in a February 13 conference call). The staff notes that DCD Section 19E.4.8 discusses the loss of coolant accident (LOCA) analysis at Mode 3 conditions and the loss the RNS events during shutdown modes. No specific information in DCD Section 19E.4.8 is related to the loss inventory aspects of the inadvertent opening of valves in shutdown modes.

Provide the exact information that is intended to be used for completion of the DCD Section 19E.4.7.1 discussion.

Westinghouse Response:

DCD 19.E.4.7.1 Revision 3 will be revised as shown.

Design Control Document (DCD) Revision:

19E.4.7.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the Automatic Depressurization System

Subsection 15.6.1 includes analyses and evaluations of the inadvertent opening of a pressurizer safety valve or the inadvertent operation of the automatic depressurization system (ADS). ~~The analyses discussed here and in subsection 15.6.1 evaluate the RCS depressurization aspect following these events. Loss of RCS inventory aspects of these events is covered in subsection 19E.4.8.2 of this appendix.~~

When analyzed as depressurization events, inadvertent opening of primary side relief valves, if the reactor is at-power, could result in exceeding core design limits, specifically DNB criteria. Violation of DNB criteria is not a realistic concern if the reactor is in any of the subcritical modes. Therefore, these events are analyzed in Mode 1 at the maximum rated power and the analysis performed bounds cases initiated from Mode 2. **These events bound events that can occur at shutdown.**

The inadvertent ADS is analyzed as a loss of coolant accident in Mode 1 to demonstrate acceptance to the limits specified 10 CFR 50.46. As described in subsection 15.6.5, this analysis is a "no-break" small-break LOCA calculation. The inadvertent opening of the 4-inch nominal size ADS Stage 1 valves is a situation that minimizes the venting capability of the reactor coolant system. Only the ADS valve vent area is available; no additional vent area exists due to a break. This case examines whether sufficient vent area is available to completely depressurize the reactor coolant system and achieve injection from the IRWST without core uncovery. The case analyzed at power bounds the inadvertent ADS during shutdown, because the lower decay heat levels at shutdown reduce the challenge to the ADS vent capacity. More limiting loss of coolant accidents at shutdown are analyzed as described in DCD section 19.E.4.8.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.028 (Response Revision 1)

Question:

(Section 16.1, TS 3.4.12 Action A and TS 3.4.13 Action A) Condition A lists three possible conditions. Does each condition get a separate Completion Time? How does this action requirement work? None of the examples in Section 1.3 address this format. Please describe a specific example for each Specification. There may be a clearer way to render these action statement conditions.

Westinghouse Response:

TS 3.4.12 Action A lists three separate Conditions that each have their own 72 hour Completion Time to return the affected flowpath(s) to operable status. TS 3.4.13 Action A lists two separate Conditions that each have their own 72 hour Completion Time to return the affected flowpath(s) to operable status.

The application of the Completion Times as described above for the multiple conditions within a Condition Statement in TS 3.4-12 and TS 3.4-13 (as well as the identified STS TSs above) is consistent with the NRC discussion for the application of Completion Times for this type of Condition Statement as described by the NRC reviewer in Item A of RAI 630.039.

The format of these Actions are consistent with the format of the following STS, Rev. 2 Action statements: TS 3.3.6 Action B; TS 3.4.12 Action G; TS 3.4.16 Action C; TS 3.5.4 Action A.

Since the format is consistent with formats used in the STS, Rev. 2, no change to the AP1000 TS or addition to the examples in Section 1.3 is deemed necessary, in order to prevent deviation from the STS Rev. 2 where there is no technical difference due to the AP1000 plant design.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

NRC Additional Comments:

Condition A for these Technical Specifications still require further clarification.

Westinghouse Additional Response:

TS 3.4.12 and 3.4.13 have been re-numbered as TS 3.4.11 and 3.4.12, respectively, in Rev. 3 of the AP1000 Technical Specifications.

During discussions with the NRC reviewer on this response, there is need for further clarification so that there is no confusion about the possibility of having an Action statement that could allow both ADS stage 1 and stage 2 to be inoperable, while simultaneously having another ADS stage 1 and stage 3 inoperable.

The Actions for these two Technical Specifications have been revised to separate the single path inoperable and two paths inoperable into two different Conditions and Action statements. The Action statement for two flow paths inoperable has been revised to use a logical AND statement and clarify that "either a stage 2 or 3 path" is inoperable at the same time as a stage 1 path. This eliminates any potential confusion about simultaneously having stage 2 and stage 3 paths inoperable.

The change to the TS Action statements and the associated Bases discussion for these Actions have been revised as shown.

DCD Revision 3 incorporates the changes identified in the original response to this RAI. DCD Revision 3 will be updated to address this change.

Design Control Document (DCD) Revision:

DCD TS 3.4.11, Revision 3 (previously TS 3.4.12) Actions will be revised as follows:

ACTIONS				
CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One flow path inoperable. ~~OR~~	A.1	Restore flow path(s) to OPERABLE status.	72 hours
B.	One stage 1 ADS flow path inoperable~~ and one stage 2 ADS flow path inoperable~~. **AND**~~OR~~ **Either** ~~O~~one stage **2**~~1~~ ADS flow path inoperable **or**~~and~~ one stage 3 ADS flow path inoperable.	**B.1**	**Restore flow path(s) to OPERABLE status.**	**72 hours**
C. ~~B.~~	Required Action and associated Completion Time not met. OR Requirements of LCO not met for reasons other than Condition A.	**C**~~B~~.1 AND **C**~~B~~.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

DCD TS 3.4.12, Revision 3 (previously TS 3.4.13) Actions will be revised as follows:

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One required flow path inoperable. ~~OR~~	A.1	Restore flow path(s) to OPERABLE status.	72 hours
B.	One required stage 1 ADS flow path **inoperable.** **AND** ~~and~~ **Either** one required stage 2 or stage 3 ADS flow path inoperable.	**B.1**	**Restore flow path(s) to OPERABLE status.**	**72 hours**
C. ~~B.~~	Required Action and associated Completion Time not met. OR Requirements of LCO not met for reasons other than Condition A.	**C**~~B~~.1	Initiate action to be in MODE 5, with RCS open and ≥ 20% pressurizer level.	Immediately

DCD TS 3.4.11, Revision 3 (previously TS 3.4.12) Bases Actions will be revised as follows:

ACTIONS	A.1 If any one~~, or if two~~ flow path~~s, consisting of one stage 1 and one stage 2 or 3, are~~ **is** determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function as long as a single failure does not also occur. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a DBA during this time period. This Completion Time is the same as is used for two train ECCS systems which are capable of performing their safety function without a single failure.
	B.1 **If two flow paths, consisting of one stage 1 and either one stage 2 or 3, are determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function as long as a single failure does not also occur. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a DBA during this time period. This Completion Time is the same as is used for two train ECCS systems which are capable of performing their safety function without a single failure.**
	~~C~~B.1 and ~~C~~B.2 If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.11 are not met for reasons other than Condition A, the plant must be brought to MODE 5 where the probability and consequences on an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

DCD TS 3.4.12, Revision 3 (previously TS 3.4.13) Bases Actions will be revised as follows:

ACTIONS	A.1 If any one~~, or if two~~ flow path~~s, consisting of one stage 1 and one stage 2 or 3, are~~ **is** determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is acceptable since the OPERABLE ADS paths can mitigate shutdown events without a single failure.
	B.1 **If two flow paths, consisting of one stage 1 and either one stage 2 or 3, are determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is acceptable since the OPERABLE ADS paths can mitigate shutdown events without a single failure.**
	~~C~~B.1 If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.12 are not met for reasons other than Condition A, the plant must be placed in a MODE in which this LCO does not apply. Action must be initiated, immediately, to place the plant in MODE 5 with the RCS pressure boundary open and ≥ 20% pressurizer level.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.052 (Response Revision 1)

Question:

(Section 16.1, TS Section 3.9) STS Section 3.9 does not contain specifications for a decay time limit and a spent fuel pool fuel handling crane load limit. Nevertheless, such specifications must be included in the AP1000 TSs unless a suitable justification is provided, because these "operating restrictions" are directly related to assumptions of the design-basis fuel handling accident in the spent fuel storage pool, and satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). Provide design-specific justifications for omitting these LCOs from the AP1000 TSs.

Westinghouse Response:

The AP1000 technical specifications do not include a specification on decay time limit because the doses resulting from a fuel handling accident in the spent fuel handling area are not sensitive to the decay time assumed in the analysis. In the AP1000, a time of 100 hours after shutdown was assumed for the time of the accident to occur. The resulting doses are less than 25% of the dose guideline identified in 10 CFR Part 50.34. A sensitivity study was performed in which the fuel handling accident was assumed to occur just 24 hours after shutdown, even though this is an unrealistically short time for fuel movement. The resulting doses are still less than the guideline in 10 CFR Part 50.34. Therefore, the results of the fuel handling accident analysis are not sensitive to the decay time assumed for the dropped assembly, and thus a technical specification is not necessary to ensure safe fuel handling operations.

The AP1000 technical specifications do not include a specification on spent fuel pool fuel handling crane load limit because the AP1000 spent fuel handling machine is limited to lifting only one fuel assembly at a time. The fuel handling machine, which is the only machine to handle spent fuel in the spent fuel storage area, lifts each fuel assembly into a mast before it can be moved. The design of the machine allows only one assembly in the mast. Therefore, by design, the maximum load the machine will lift is limited to one fuel assembly plus handling tools. Since the spent fuel handling machine is limited to lifting one fuel assembly, and the fuel handling accident assumes one dropped fuel assembly, a technical specification on the fuel handling machine load limit is not necessary to ensure safe fuel handling operations.

In previous NRC reviews of other plant designs, Refueling Operations Technical Specifications without a decay time limit and a spent fuel pool fuel handling crane load limit were found acceptable. Both the AP600 and the System 80+ received design certification without including these two technical specifications.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

The discussion of the decay time is insufficient - a 100 hour assumption would likely not be short enough to use the "physically impossible" argument. In order to not include a technical specification LCO for the decay time, the fuel handling accident analysis must assume a decay time that is clearly less than the time physically needed to begin moving fuel assemblies out of the core following unit shutdown for refueling, and that analysis should predict acceptable dose consequences. Staff does not consider 100 hours to be short enough.

Westinghouse Additional Response:

The AP1000 Technical Specifications do not require an LCO decay time limit.

The limiting dose resulting from a design basis fuel handling accident is the dose at the site boundary. Assuming a decay time of 100 hours for the analysis of a design basis fuel handling accident, the resulting site boundary dose is less than 25% of (well within) the dose guideline identified in 10 CFR 50.34.

An evaluation was also performed assuming a decay time of only 24 hours. Although 24 hours is an unrealistically short time to establish the required plant conditions for movement of the fuel or upper internals, the resulting doses for a design basis accident with a decay time of 24 hours are still less than 25% of the guideline in 10 CFR 50.34. Therefore, the safety analyses show that the capability of the AP1000 design to meet the dose acceptance criterion of 6.3 rem TEDE (Regulatory Guide 1.183) for the fuel handling accident is not sensitive to the decay time assumed for the dropped fuel assembly.

Since no operating restrictions are required as an initial condition of the design basis accident, in accordance with Criterion 2 of 10 CFR 50.36, a technical specification on decay time is not necessary to ensure safe fuel handling operations.

The discussion of the AP1000 offsite doses for the fuel handling accident described in DCD 15.7.4.5 will be revised to also provide the calculated doses from an evaluation of a fuel handling accident, assuming a decay time of 24 hours. The discussion confirms the relative insensitivity to decay time for meeting the dose limits and precludes the need for any operating restrictions on decay time that would require a Technical Specification.

DCD Revision 3 incorporates the changes identified in the original response to this RAI. DCD Revision 3 will be updated to address this change.

Design Control Document (DCD) Revision:

The following paragraph will be added to DCD 15.7.4.5:

15.7.4.5 Offsite Doses

Using the assumptions from Table 15.7-1, the calculated doses from the initial releases are determined to be 2.4 rem TEDE at the site boundary and 0.6 rem TEDE at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "well within" is taken as meaning 25 percent or less.

Additionally, an evaluation was performed to determine the impact of a fuel handling accident occurring with less than 100 hours of fission product decay. The evaluation assumed that the fuel handling accident occurred after only 24 hours of decay. The resulting doses are <4.3 rem TEDE at the site boundary and <1.0 rem TEDE at the low population zone outer boundary. These doses remain well within the dose guideline of 25 rem TEDE.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.013 (Response Revision 1)

Question:

For post ADS long-term cooling, Section A3.5 (and Section A5.1 Long-Term Cooling Cases No. 5 and 6) provides your judgement, based on AP1000's increased power level, ADS4 flow capacity, IRWST injection, and containment recirculation over the AP600 design. The conclusion provided is that the AP1000 ADS4 vent capacity is sufficient for justification of the long-term core cooling success criteria assumed in the AP1000 PRA. Section A5.2 states that Section A5.5.2 documents the long-term cooling analyses performed with the WCOBRA/TRAC code with the details of the analysis methodologies used provided within each subsection. However, there is no Section A5.5.2.

Provide the WCOBRA/TRAC long-term cooling analysis as part of AP1000 PRA to support your conclusion.

Additional Question:

A. For long-term cooling Cases F and G, the availability of 1/1 CMT (A) is assumed. Are these cases bounding the 6 cases, listed in Table 3-3 and 3-4 in Attachment 1 to RAI 720-012, where both CMTs fail?

B. Section 1 states that it is conservatively assumed that the DEDVI line break occurs in the PXS-B room, since the size of this room is bigger than PXS-A. In Section 4.1, 3rd paragraph, it is stated that "initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Fig. 720.013-14). Since the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from downcomer is vented out through the break (Fig. RAI 720.023-13[TYPO]." However, Figure 720.013-14 shows DVI-B mixture flow rate, and Figure 720-013-13 shows DVI-A mixture flow rate. There appears to be discrepancies on the break location. Clarify whether the break is in DVI-A or DVI-B.

C. Same question for Section 4.2, Figures 720.013-14 and -13.

D. WCOBRA/TRAC boundary conditions including the interaction between the AP1000 primary system, passive safety systems, containment, and containment systems are stated to be calculated with MAAP. The NRC staff has not reviewed and approved the MAAP code for this purpose. (See open issues involving RAI 720.021 below). Furthermore in the response to RAI 440.009 (dated Sept 12, 2001) Westinghouse indicated that MAAP would not be used with WCOBRA/TRAC for long-term cooling analysis. This issue will remain open until Westinghouse provides long term uncertainty analyses using approved methodology.

Westinghouse Response to Additional Questions: < Note that the original (unchanged) response is provided afterwards. >

A. In the two LTC T&H uncertainty cases (F and G), 1 CMT and 0 accum are assumed to inject. The other CMT and both accum are all assumed to remain filled with water, so that they don't contribute to the containment water levels during long term cooling. There is no effect on the long term cooling if 1 CMT or 1 accum injects, because the mass of water in the containment is completely dominated by the IRWST volume and the difference in CMT and accum water volume is insignificant. Therefore, the cases analyzed with 1 CMT bound cases with 1 accum for LTC.

B. The event analyzed was a DVI B LOCA. The plots for both cases F and G were miss labeled. The plots in PRA Appendix A, rev 1, will be revised.

C. See response to item B above.

D. The use of the MAAP code is considered reasonable in support of the PRA LTC T&H uncertainty calculations. MAAP only provides the boundary conditions used in the WCOBRA-TRAC analysis. MAAP is not used to support the DCD LTC analysis. This same approach was used and accepted by the NRC for the T&H uncertainty analysis performed for the AP600.

Design Control Document (DCD) Revision:

None

PRA Revision:

PRA Appendix A figures will be re-labeled as follows:

	Current	Revised
A5.3-13	Case F – DVI-A Mixture Flow Rate	Case F – DVI-B Mixture Flow Rate
A5.3-14	Case F – DVI-B Mixture Flow Rate	Case F – DVI-A Mixture Flow Rate
A5.3-27	Case G – DVI-A Mixture Flow Rate	Case G – DVI-B Mixture Flow Rate
A5.3-28	Case G – DVI-B Mixture Flow Rate	Case G – DVI-A Mixture Flow Rate

Westinghouse Response: < Original (unchanged) response follows. >

Attached to this RAI response is analysis of two thermal hydraulic long-term cooling (LTC) analysis cases. These cases were determined to be the low margin, risk important cases that should be analyzed to bound their T/H uncertainty (refer to response to RAI 720.012). This WCOBRA-TRAC analysis shows that adequate core cooling is provided for these cases.

Design Control Document (DCD) Revision:

None

PRA Revision:

The attached analysis will be appropriately inserted into Appendix A of the PRA.

1. Objective

The objective of these analyses is to analyze the AP1000 long-term core cooling (LTCC) behavior following a guillotine double-ended direct vessel injection (DEDVI) line break to support the PRA thermal hydraulic (T/H) uncertainty evaluations. In order to bound the T&H uncertainty, this analysis is performed using the DCD code and conservative methods.

Two cases of LTCC following a DEDVI line break are analyzed. These cases were determined by T/H uncertainty evaluations performed for AP1000 (refer to RAI 720.012). One of these cases considers that the containment is isolated (case F), and the other case considers that the containment isolation has failed (case G). It is conservatively assumed that the DEDVI line break occurs in the PXS-B room. Since the size of this room is bigger than PXS-A, it reduces the containment water level during recirculation. It also takes more time for the water to fill it to the DVI nozzle elevation, where water can start flowing into the downcomer through the broken DVI line. In both cases, the general assumptions and methodology of the calculations are essentially the same. Conservative boundary and initial conditions are applied consistent with these multiple failure PRA based scenarios to ensure that the thermal/hydraulic uncertainties contained within the success criteria are bounded.

Below is a short summary of the two T/H uncertainty cases being described herein.

Case F:

- DEDVI LOCA in line B
- Available equipment - 1/1 CMT (A), both IRWST injection lines open with 1/2 valves open in each, only 1 recirculation line available with both valves open and this is the line attached to DVI-B, **3/4 ADS-4**, PCS water drain with 1/3 valves open
- Unavailable equipment - no ADS 1/2/3, PRHR, RNS injection / spill, IRWST gutter
- **Containment isolation** is assumed to have **worked.**

Case G:

- DEDVI LOCA in line B
- Available equipment - 1/1 CMT (A), both IRWST injection lines open with 1/2 valves open in each, 1/2 recirculation lines open with both valves open (line B), **4/4 ADS-4**, PCS water drain with 1/3 valves open
- Unavailable equipment - no ADS 1/2/3, PRHR, RNS injection / spill, IRWST gutter
- **Containment isolation** is assumed to have **failed** (18" HVAC line remains open).

2. WCOBRA/TRAC Modeling Methodology

The simulation methodology used in the current analyses is essentially the same as the one used for the AP600 design certification process (Ref.1).

- The T/H uncertainty analyses are performed using the WCOBRA/TRAC thermal hydraulic computer code (Ref.2).
- The WCOBRA/TRAC AP1000 model is the same as the one used in the AP1000 Post-LOCA Long-term Cooling analysis (Ref.3)
- The AP1000 LTCC simulations are performed using WCOBRA/TRAC in a transient mode. The transient mode approach has been validated by the Oregon State University Tests and was used in the AP600 Design Certification (Ref. 1).
- For each case, the AP1000 initial and boundary conditions are provided by a MAAP4 calculation. MAAP4 is capable of simulating the behavior and the interaction between the AP1000 primary system, the passive safety systems, containment, and the containment systems – a feature that is not present in WCOBRA/TRAC. The response to RAI 720.021 discusses the ability of MAAP4 to model the AP1000 in cases where there is a failure to isolate the containment.
- Like the MAAP4, the WCOBRA/TRAC simulation is performed with the following conservative general assumptions:
  - 102 percent core power
  - Appendix K decay heat
  - Maximum hydraulic resistance of the passive safety systems

3. Methodology Implementation

The transient mode calculation using WCOBRA/TRAC allows simulation of long transients with reasonable computer resources. As was shown in the validation of methods used in the DCD analysis (Ref.3), the calculation may be initiated from an arbitrary set of initial conditions. After an initial period of 500 to 1000 seconds the plant reaches a quasi-steady state that depends mostly on the system boundary conditions. During this "steady state" period, the boundary conditions are kept constant. After that, they are set as a function of time depending on the time window being simulated.

For the AP1000 T/H uncertainty analysis, a transient mode calculation was performed for Case F and Case G within the time period that was covered by the MAAP4 calculations for those cases. It was observed that WCOBRA/TRAC predicts higher ADS Stage 4 flows resulting in better depressurization of the primary system. Consequently, the predicted IRWST injection rates were higher when using WCOBRA/TRAC. Because of the faster IRWST draining it was estimated that the IRWST would reach its lowest level about 2 hours earlier than as predicted by MAAP4.

For each of the cases analyzed here (Case F and Case G), the IRWST level calculated by MAAP4 was adjusted to account for the more rapid draining predicted by WCOBRA/TRAC. The adjusted IRWST levels were then used as boundary conditions for each of the cases F and G.

The containment pressure, PXS-B level, IRWST and PXS-B temperatures calculated by MAAP4, together with the adjusted IRWST level, were used to define the limiting conditions that were used to assess the performance of the AP1000 passive safety system.

The following two sections document the results of the WCOBRA/TRAC simulations for these limiting windows performed for Cases F and G.

4. Results

4.1 Case F - DEDVI Line Break in the PXS-B Room with 3 of 4 ADS Stage 4, Containment Isolated

This subsection presents the simulation results of T/H uncertainty Case F – DEDVI line break located in the PXS-B room with 3 out of 4 ADS Stage 4 valves opened and the containment isolated. The initial conditions are based on the MAAP4 calculation results of the same accident scenario. They are selected such that the WCOBRA/TRAC simulation begins 3992 sec (~1 hour 6 min) after the break - shortly after IRWST injection begins.

For this transient, the initial IRWST level is 126.4 feet and its temperature is 121 deg F. The initial level in the PXS-B room is 95.8 feet. The available ADS Stage 4 paths are opened and the containment pressure is set to its initial value of 42.9 psia. Under these conditions, a 1000 second calculation is performed to ensure that the initial steady state conditions are achieved in the system. After that, the transient calculation is initiated with time-dependent boundary conditions taken from the MAAP4 calculation, but with adjusted IRWST level decrease, as discussed earlier.

Initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Figure RAI 720.013-14). Since at the beginning of the analysis, the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from downcomer is vented out through the break (Figure RAI 720.023-13). Water starts to flow back into the downcomer through the broken DVI line about 2 hours into the transient. This is the time when the level in the PXS-B room becomes high enough to provide sufficient driving head. At the onset of this event the additional amount of water supplied into the downcomer through the DVI break supplement's the IRWST injection. This leads to enhanced core cooling and momentarily, faster depressurization occurs at about 2.05 hours into the transient (see Figure RAI 720.013-11). Consequently, the IRWST injection is increased even further and as a result, the levels in the downcomer (Figures RAI 720.013-1), the reactor core (Figures RAI 720.013-2) and the upper plenum (Figures RAI 720.013-8) are also increased. The effect of this injection flow increase can also be seen on Figure RAI 720.013-4, which shows a sharp void fraction decrease in the upper half of the fuel region.

The available 3 out of 4 ADS Stage 4 valves provide enough venting capacity to assure adequate depressurization and successful performance of the passive safety systems (Figures RAI 720.013-9 and RAI 720.013-10). The fuel remains covered throughout the transient and adequate core cooling is provided to remove the decay heat. The Hot Rod cladding temperature is about 20 deg F above saturation (Figure RAI 720.013-12) and is steadily decreasing.

As the transient proceeds, the IRWST drains to a minimum of 107 feet at about 3.9 hours after the break. After that time, the level is kept constant at 107 feet, as predicted by MAAP4. The transient is terminated at about 4.2 hours after the break with the system in a continuing depressurization phase with stable DVI injection flows, and decreasing decay heat.

4.2 Case G - DEDVI Line Break in the PXS-B Room with 4 of 4 ADS Stage 4, Containment Isolation Failed

This subsection presents the simulation results of T/H uncertainty Case G – DEDVI line break located in the PXS-B room with all ADS Stage 4 valves available and with containment isolation failure. The initial conditions are based on the MAAP4 calculation results of the same accident scenario. They are selected such that the WCOBRA/TRAC simulation begins 3298 sec (~55 min) after the break - shortly after IRWST injection begins.

For this transient, the initial IRWST level is 127.9 feet and its temperature is 120.5 deg F. The initial level in the PXS-B room is 93.1 feet. All the ADS Stage 4 paths are opened and the containment pressure is set to its initial value of 17.08 psia, as calculated by MAAP4. Under these conditions, first a 1000 seconds calculation is performed so that the initial steady state is achieved in the system. After that, the transient calculation is initiated with time-dependent boundary conditions taken from the MAAP4 calculation, but with the adjusted IRWST level decrease.

Initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Figure RAI 720.013-28). Since at the beginning of the analysis, the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from downcomer is vented out through the break (Figure RAI 720.023-27). Water starts to flow back into the downcomer through the broken DVI line about 2 hours into the transient. This is the time when the level in the PXS-B room becomes high enough to provide sufficient driving head for this to happen. This time, unlike the Case F DVI break scenario, the transition into reversed injection of water through the break into the downcomer occurs a little earlier, and is somewhat softer. As a result, the increased depressurization rate observed in Case F does not occur. Still, the levels in the downcomer (Figures RAI 720.013-15), the reactor core (Figures RAI 720.013-16) and the upper plenum (Figures RAI 720.013-22) are maintained high enough by the available DVI injection.

The availability of all ADS Stage 4 valves provides enough venting capacity to assure adequate depressurization and successful performance of the passive safety systems (Figures RAI 720.013-23 and RAI 720.013-24). The fuel remains covered throughout the transient and adequate core cooling is provided to remove the decay heat. The Hot Rod cladding temperature is about 20 deg F above saturation (Figure RAI 720.013-26) and steadily decreasing.

As the transient proceeds, the IRWST drains to a minimum of 106.9 feet at about 3.7 hours after the break. After that time, the level is kept constant at 106.9 feet, as predicted by MAAP4. The transient is terminated at about 4.4 hours after the break with the system being in a phase with stable DVI injection flows, adequate ADS 4 flows, and decreasing decay heat.

5. References

1. WCAP-14800, *AP600 PRA Thermal/Hydraulic Uncertainty Evaluation for Passive System Reliability*, June 1997.

2. WCAP-12945, *Code Qualification Document for Best Estimate Analysis*, Volumes 1 through 5, Revision 1 (Westinghouse Proprietary).

3. *AP1000 Design Control Document*, Chapter 15.6.5.4C, Revision 2.

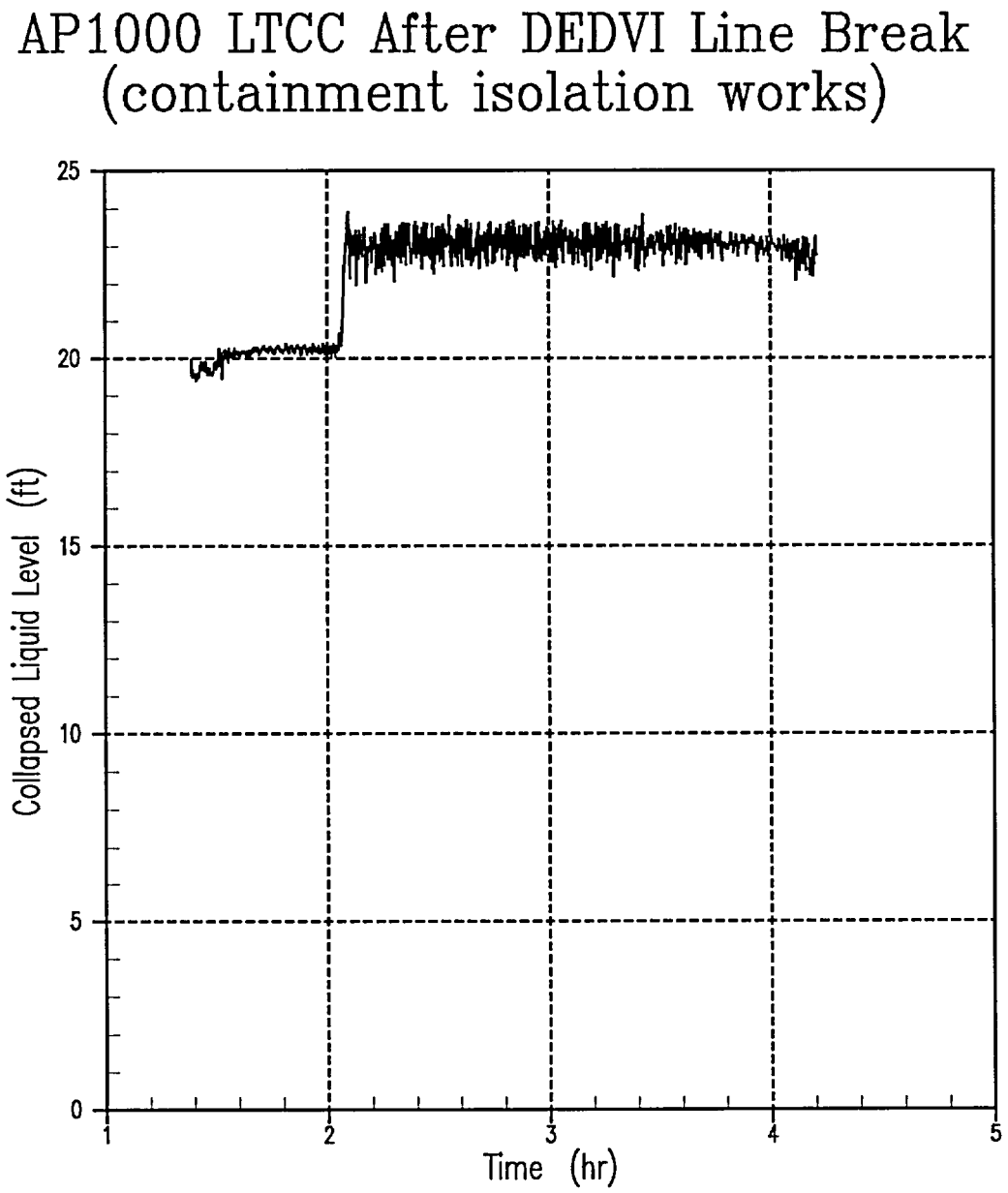


Figure RAI 720.013-1
Collapsed Level of Liquid in the Downcomer (Case F)

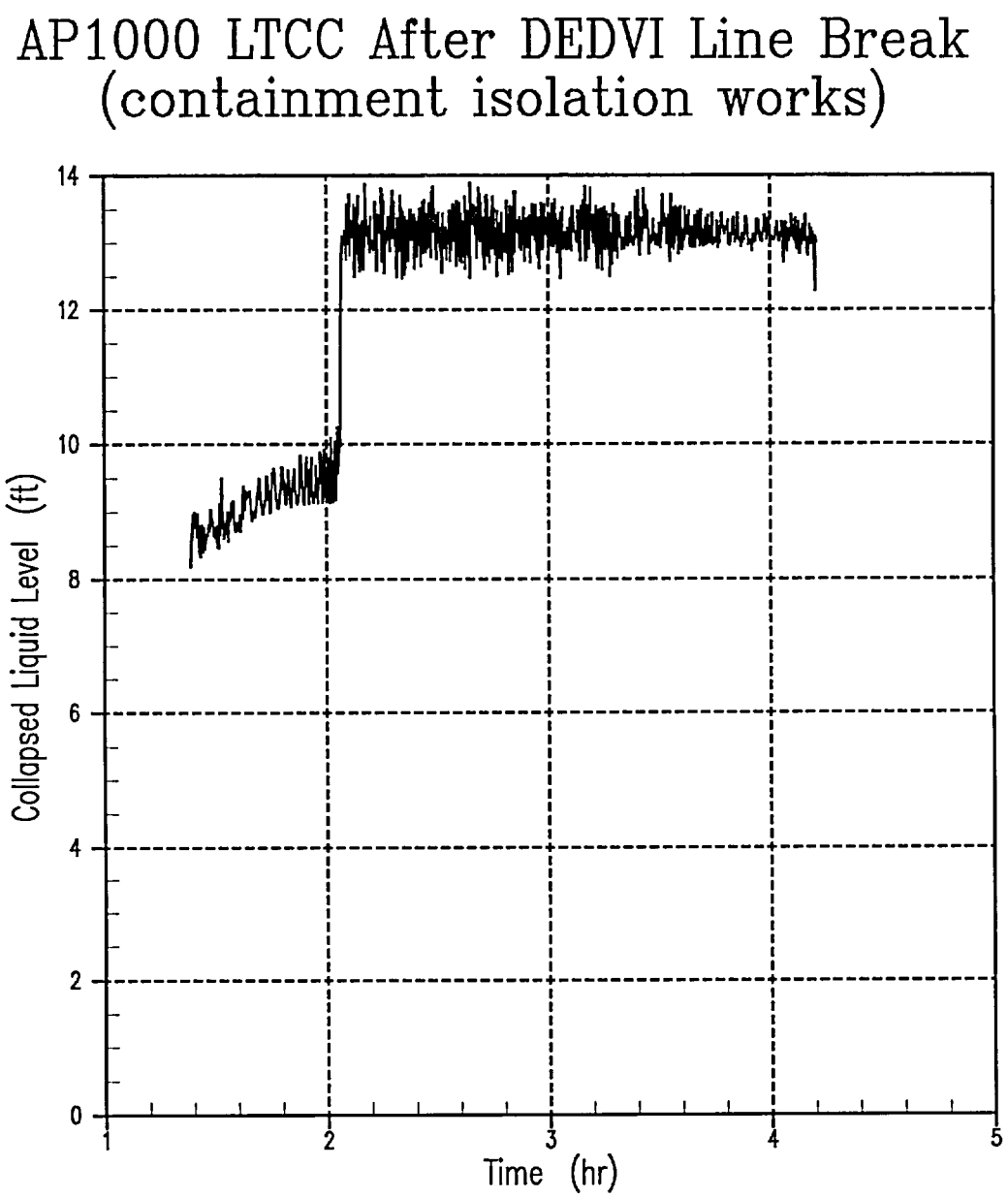


Figure RAI 720.013-2
Collapsed Level of Liquid Over the Heated Length of the Fuel (Case F)

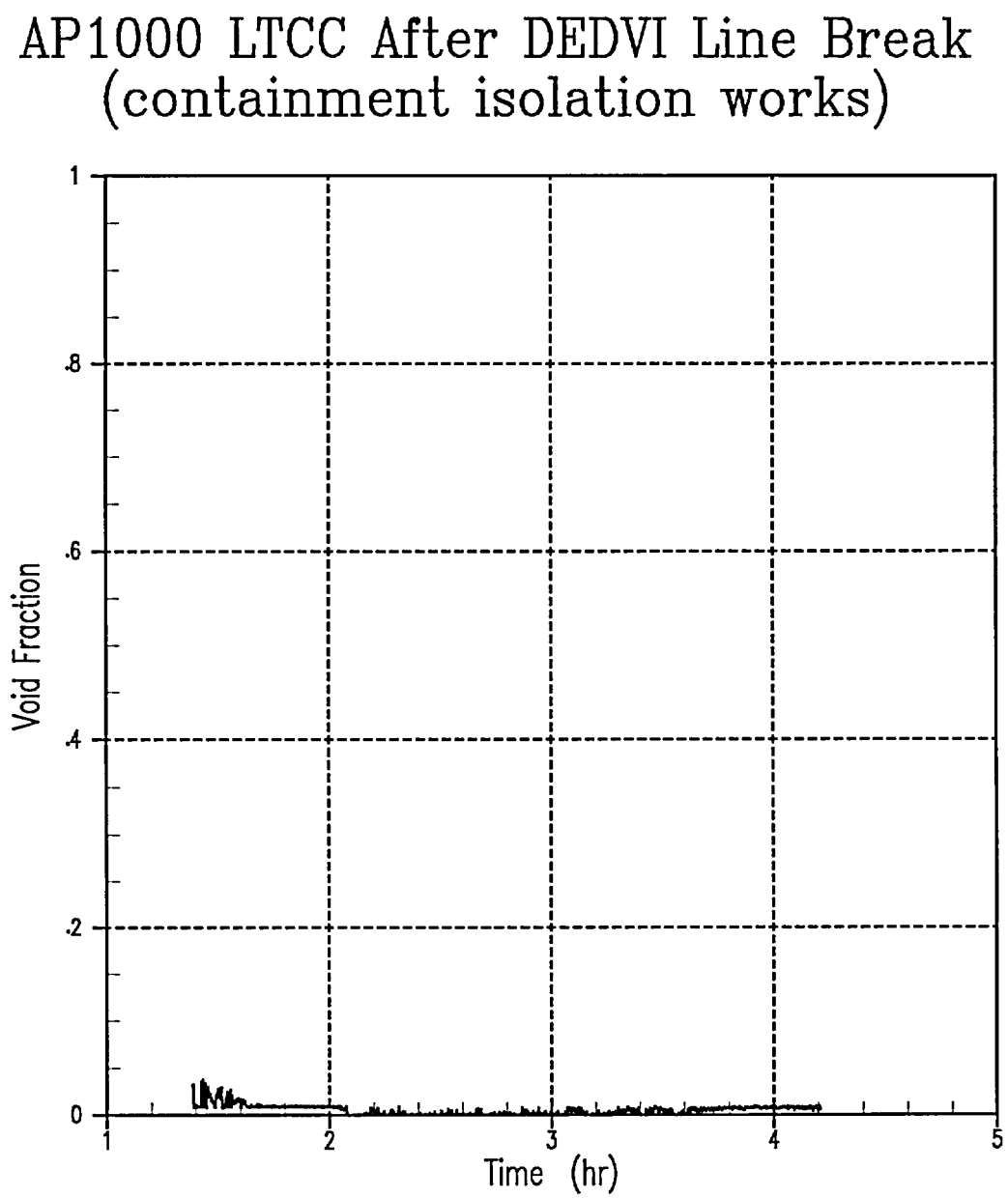


Figure RAI 720.013-3
Void Fraction in Core Cell Level 1 of 2 (Case F)

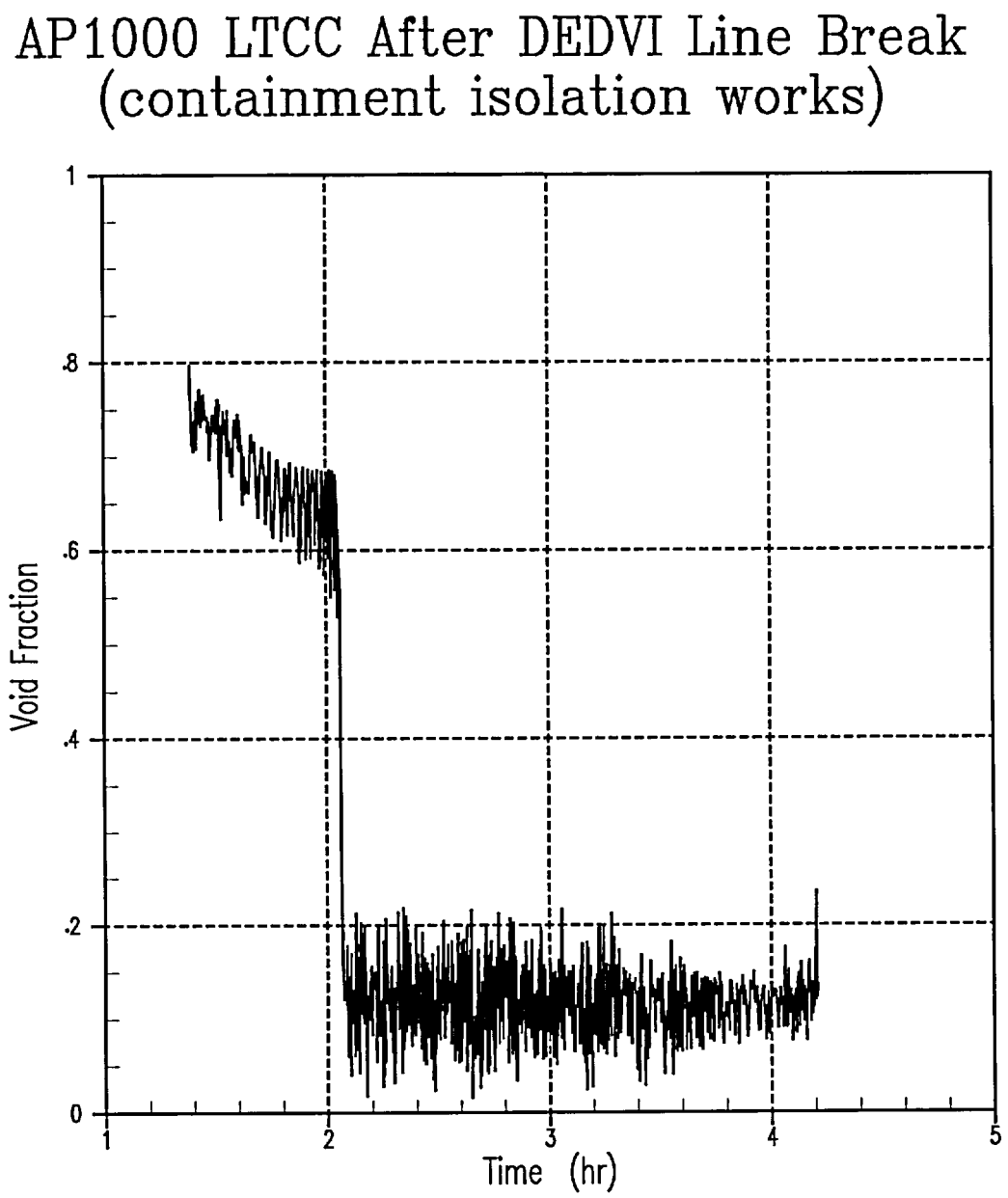


Figure RAI 720.013-4
Void Fraction in Core Cell Level 2 of 2 (Case F)

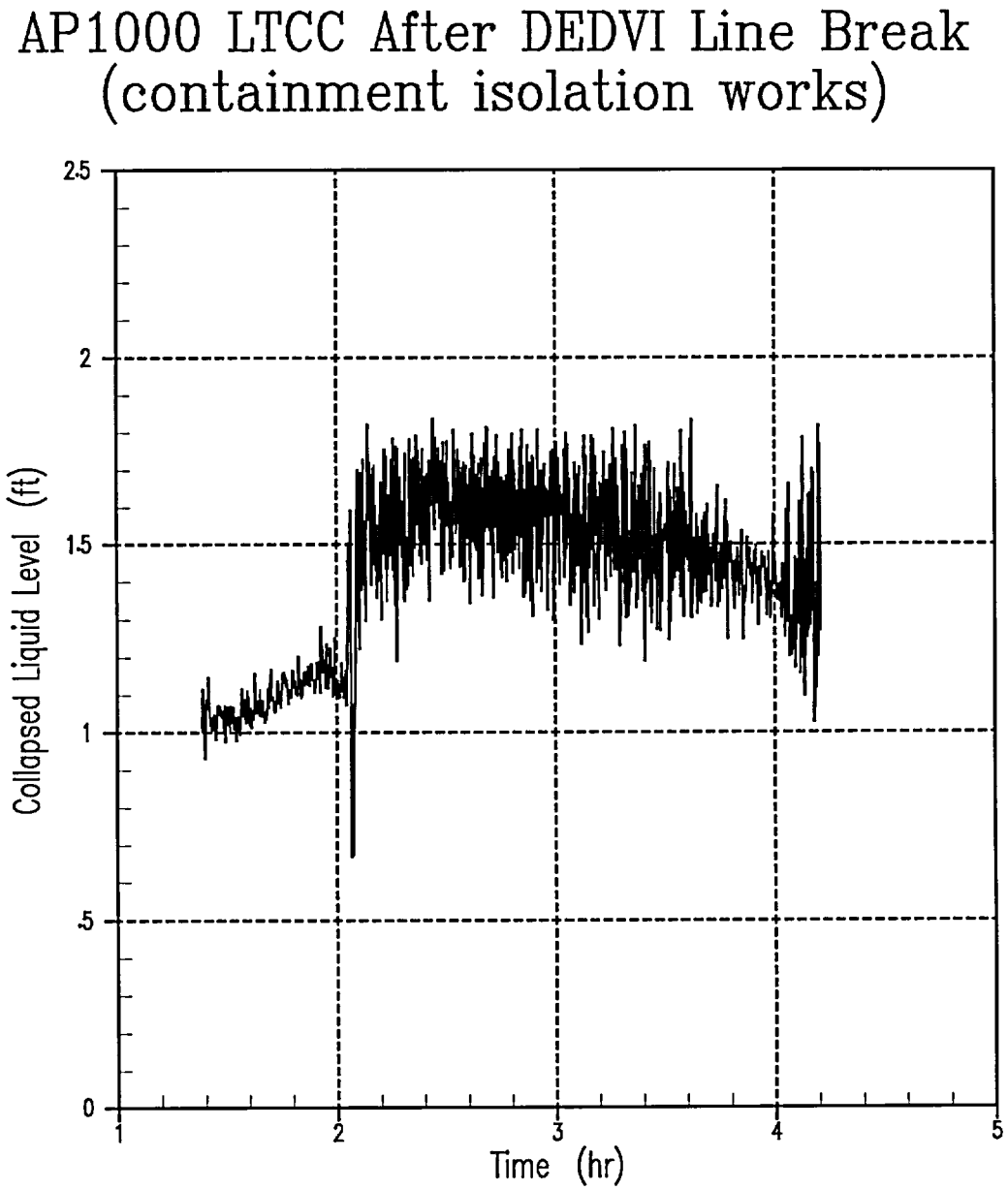


Figure RAI 720.013-5
Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (Case F)

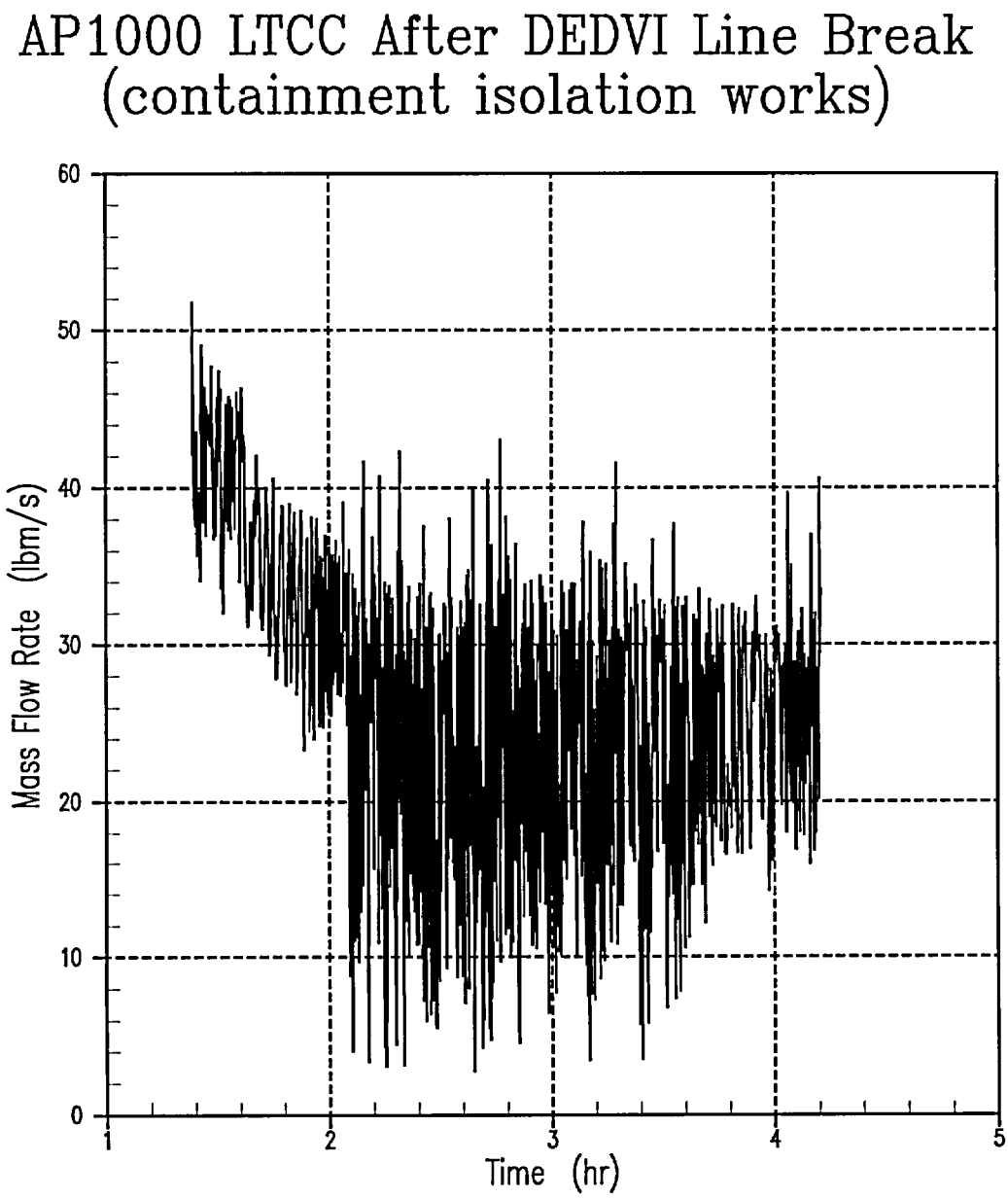


Figure RAI 720.013-6
Vapor Rate out of the Core (Case F)

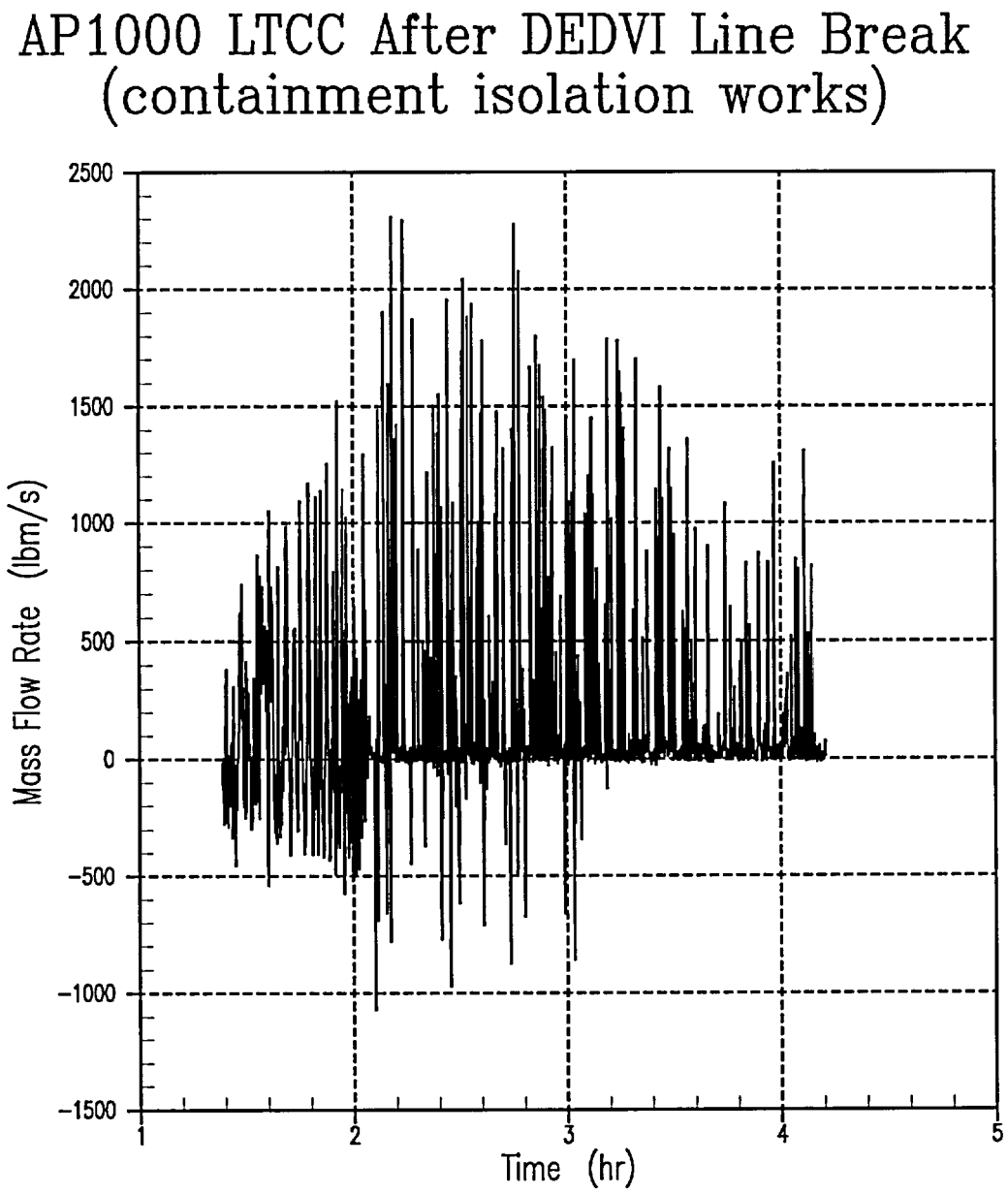


Figure RAI 720.013-7
Liquid Flow Rate Out of the Core (Case F)

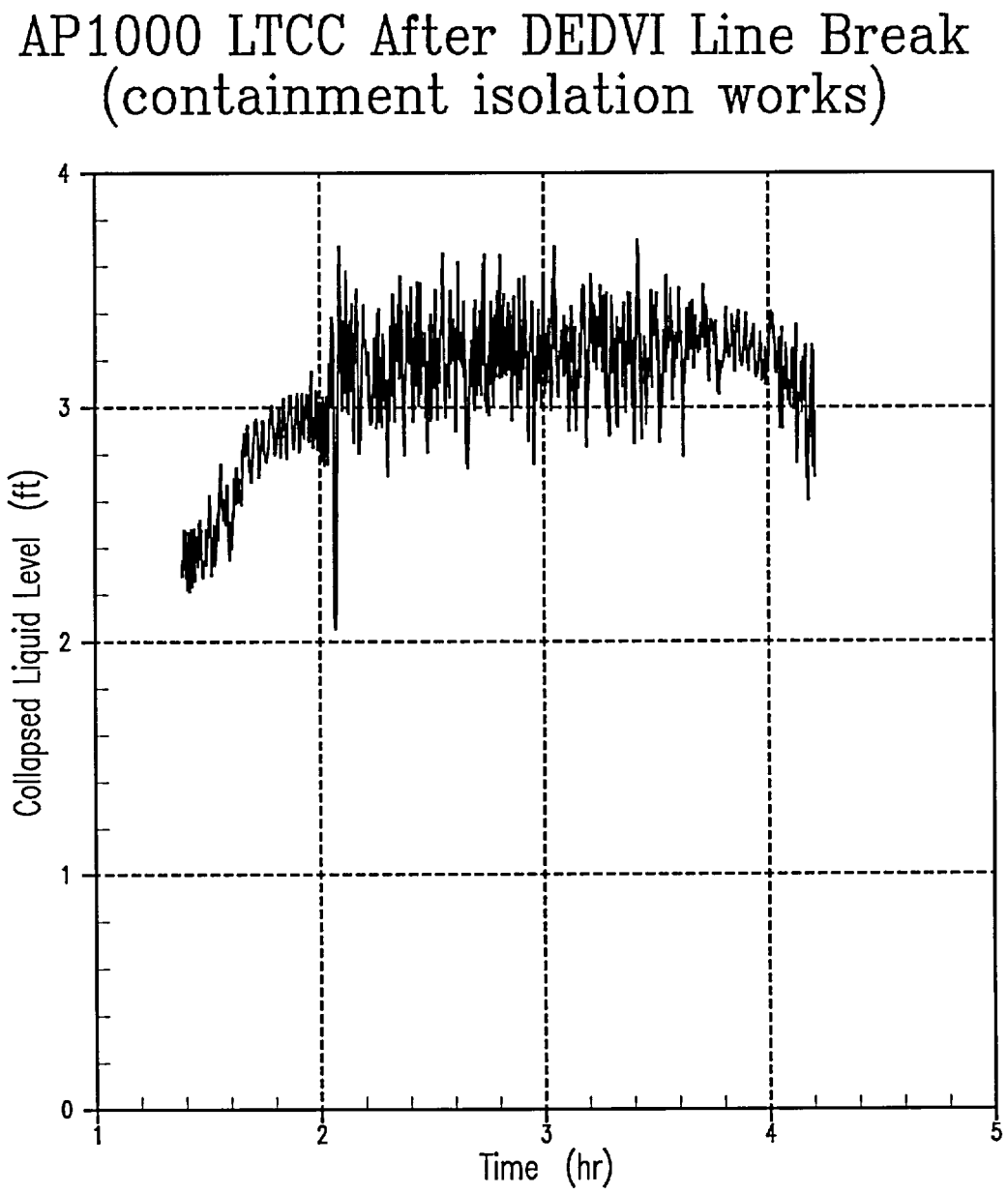


Figure RAI 720.013-8
Collapsed Liquid Level in the Upper Plenum (Case F)

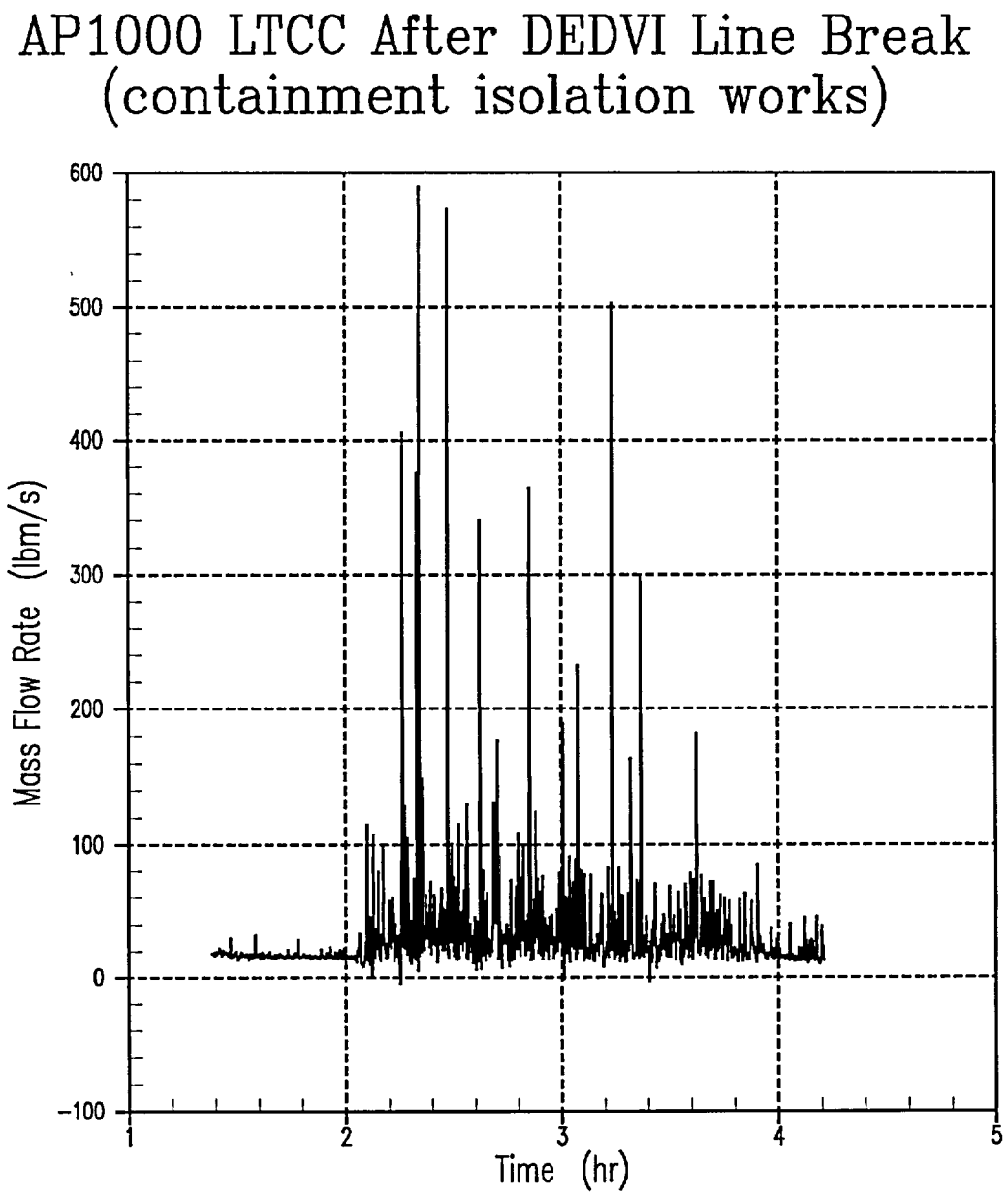


Figure RAI 720.013-9
Mixture Flowrate Through ADS Stage 4A Valves (Case F)

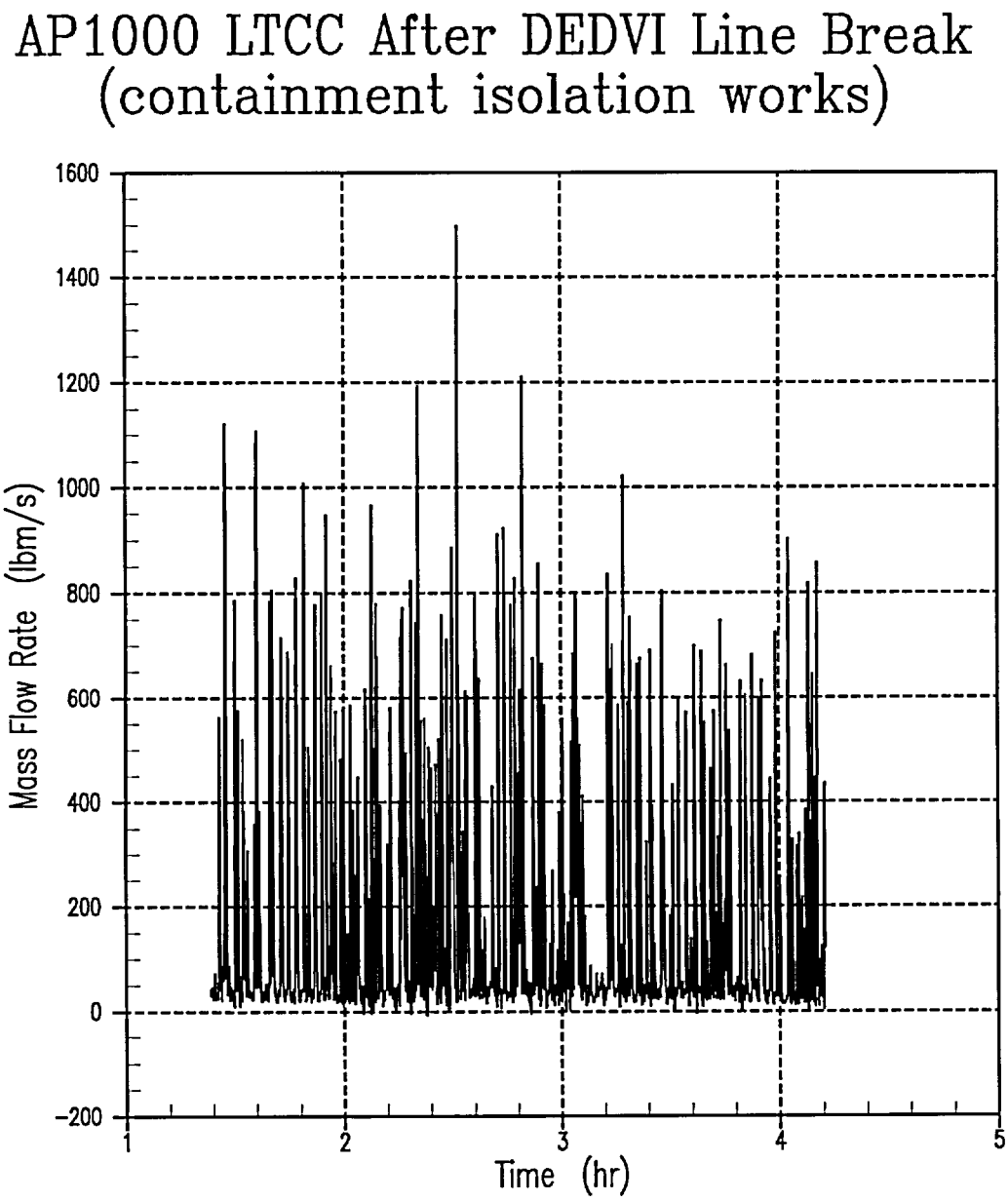


Figure RAI 720.013-10
Mixture Flowrate Through ADS Stage 4B Valves (Case F)

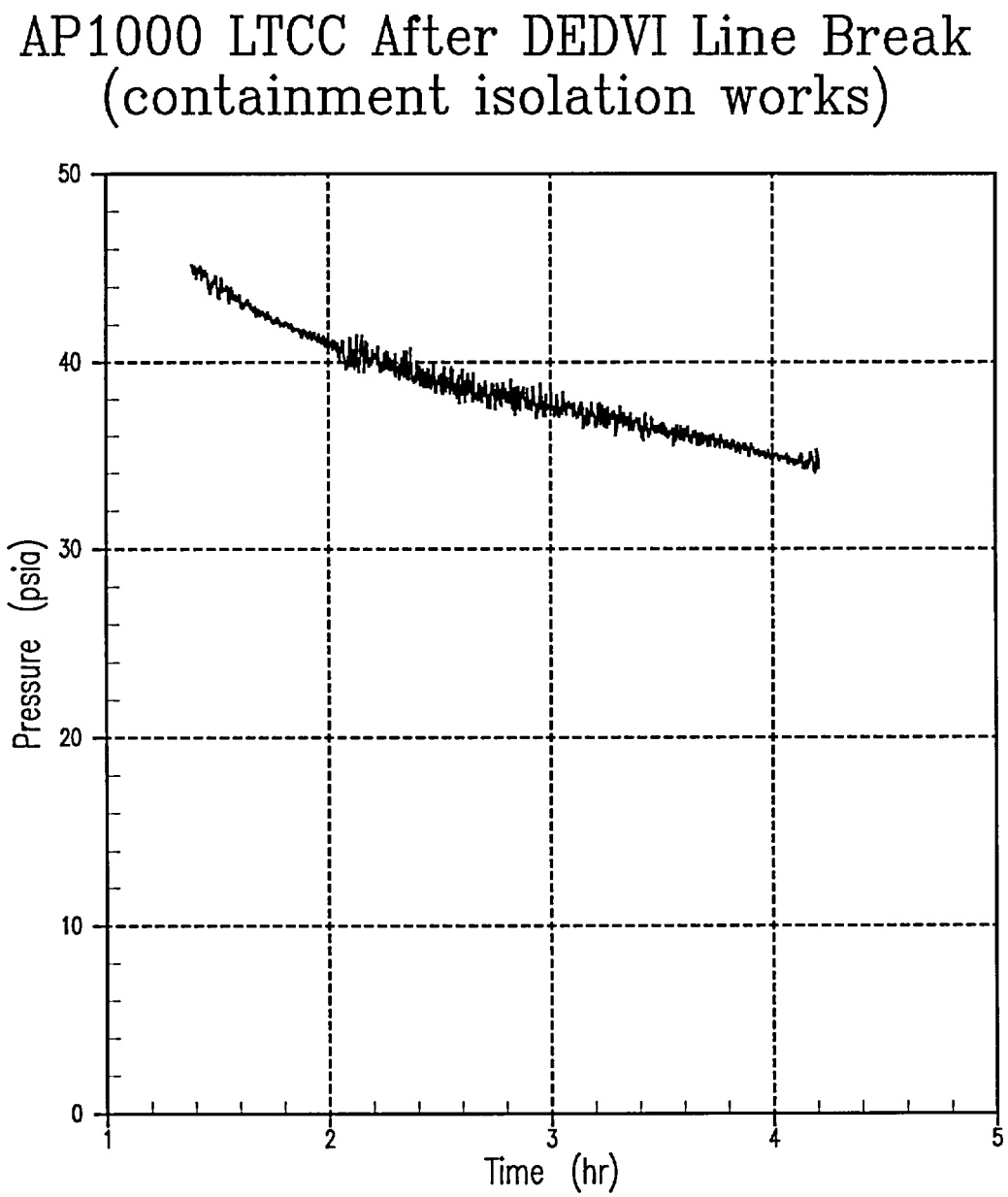


Figure RAI 720.013-11
Upper Plenum Pressure (Case F)

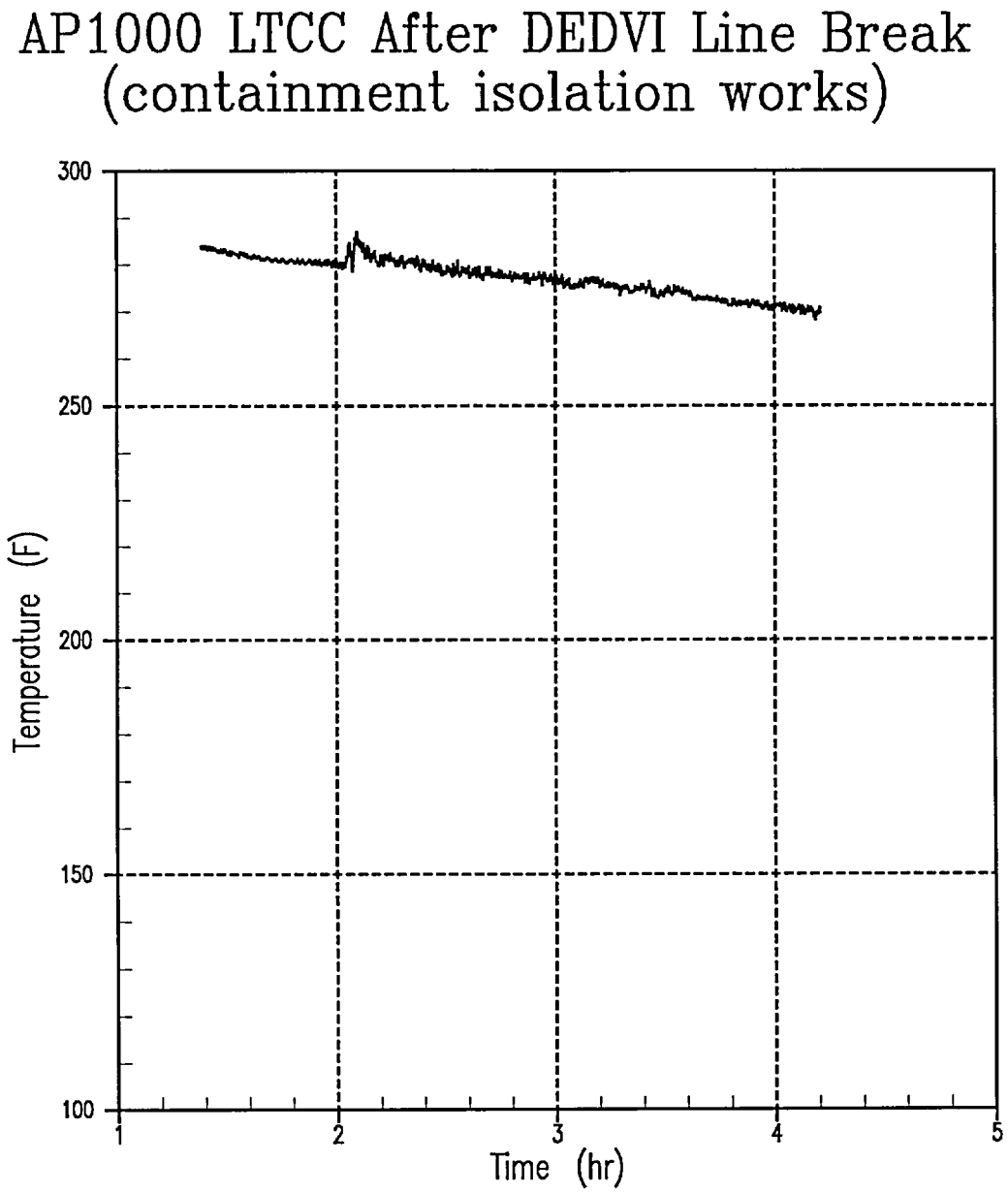


Figure RAI 720.013-12
PCT of the Hot Rod (Case F)

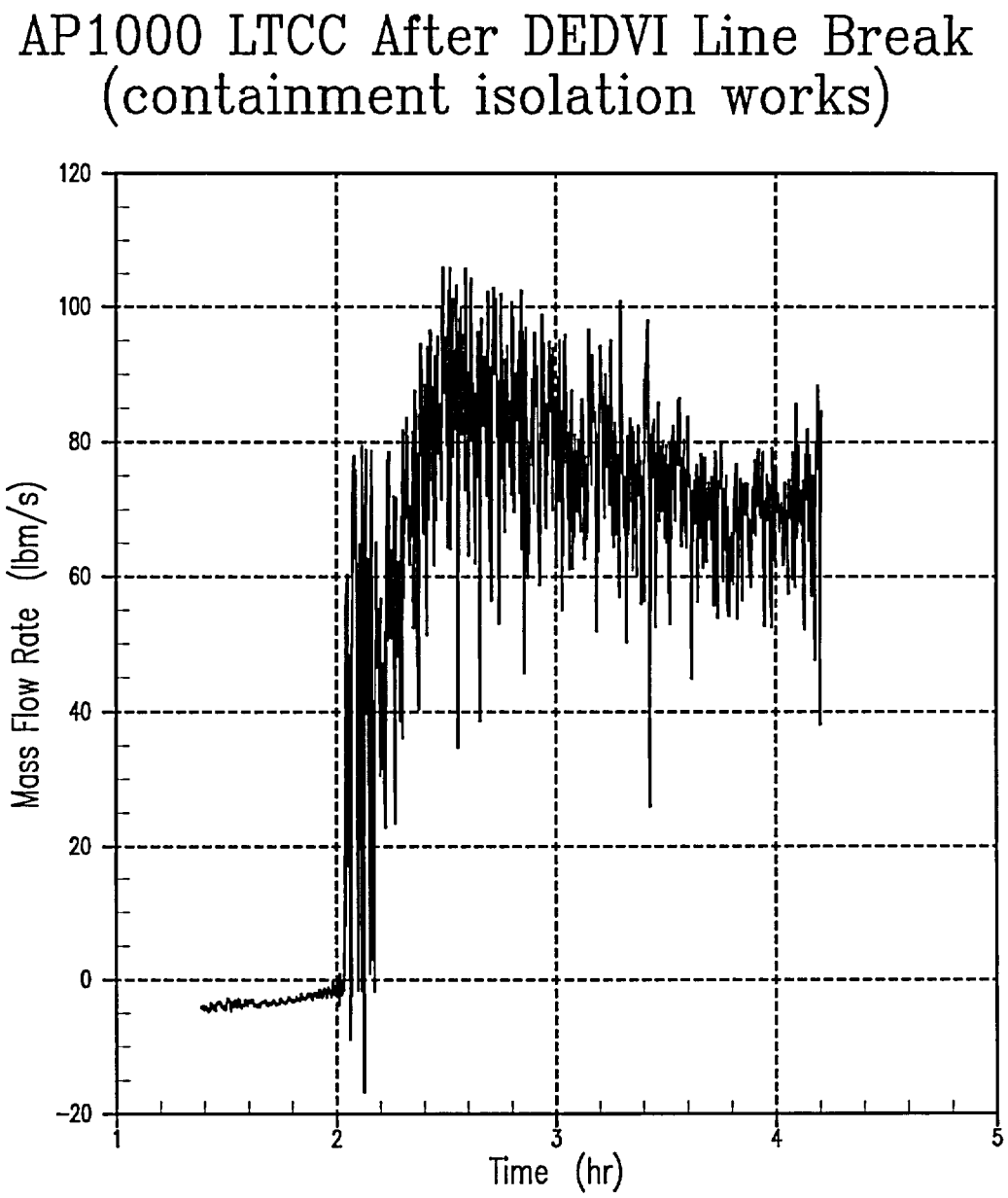


Figure RAI 720.013-13
DVI-A Mixture Flow Rate (Case F)

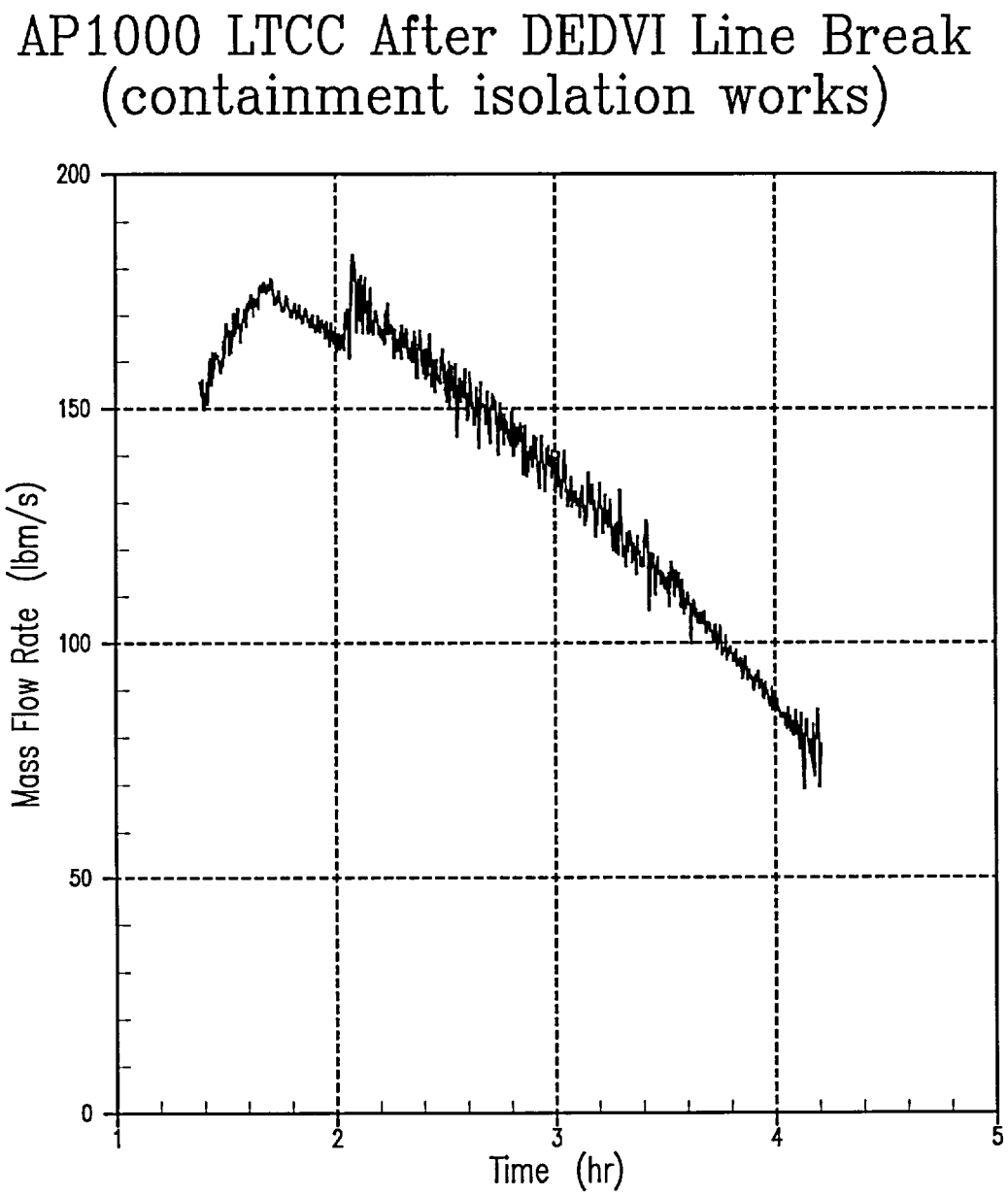


Figure RAI 720.013-14
DVI-B Mixture Flow Rate (Case F)

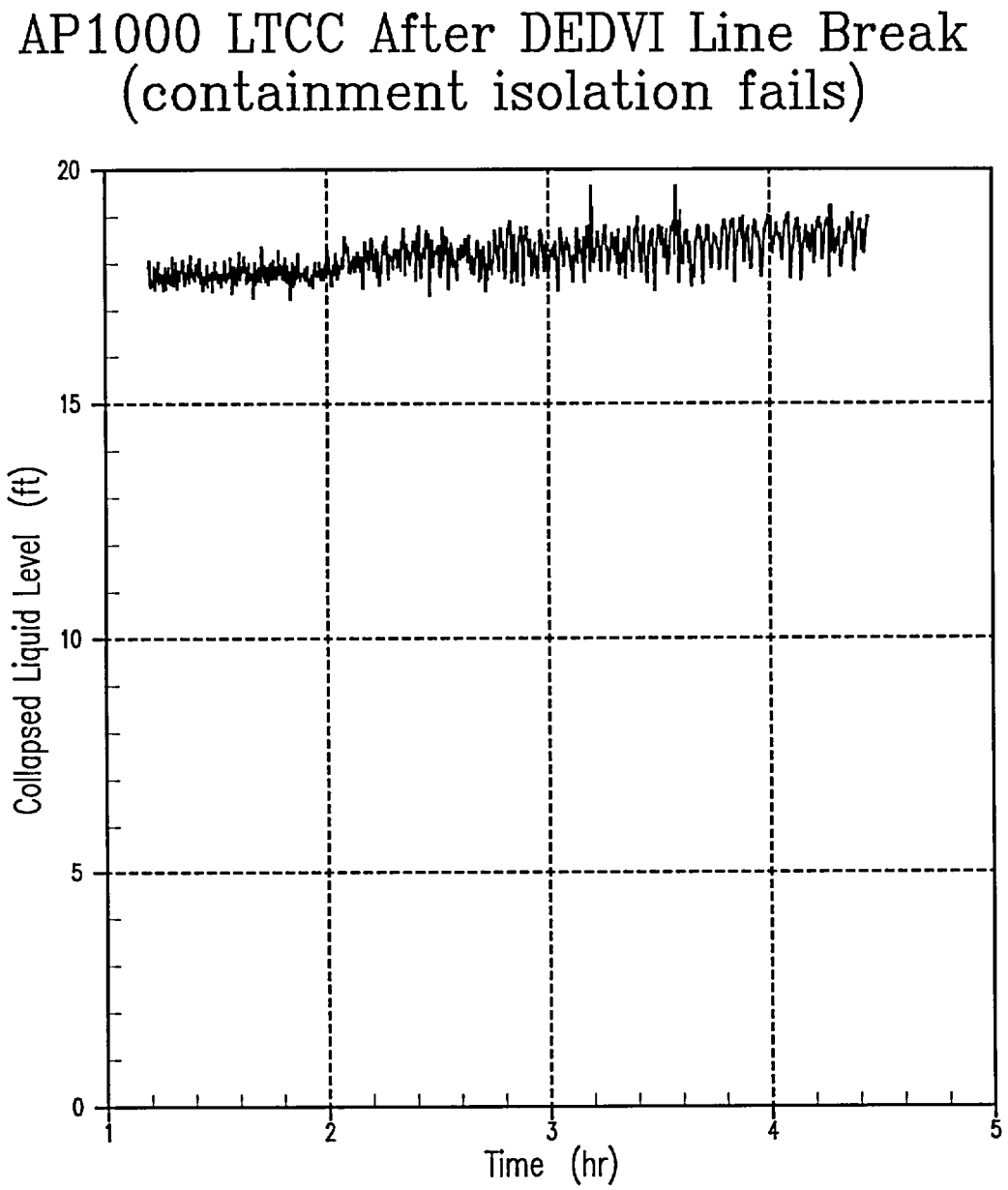


Figure RAI 720.013-15
Collapsed Level of Liquid in the Downcomer (Case G)

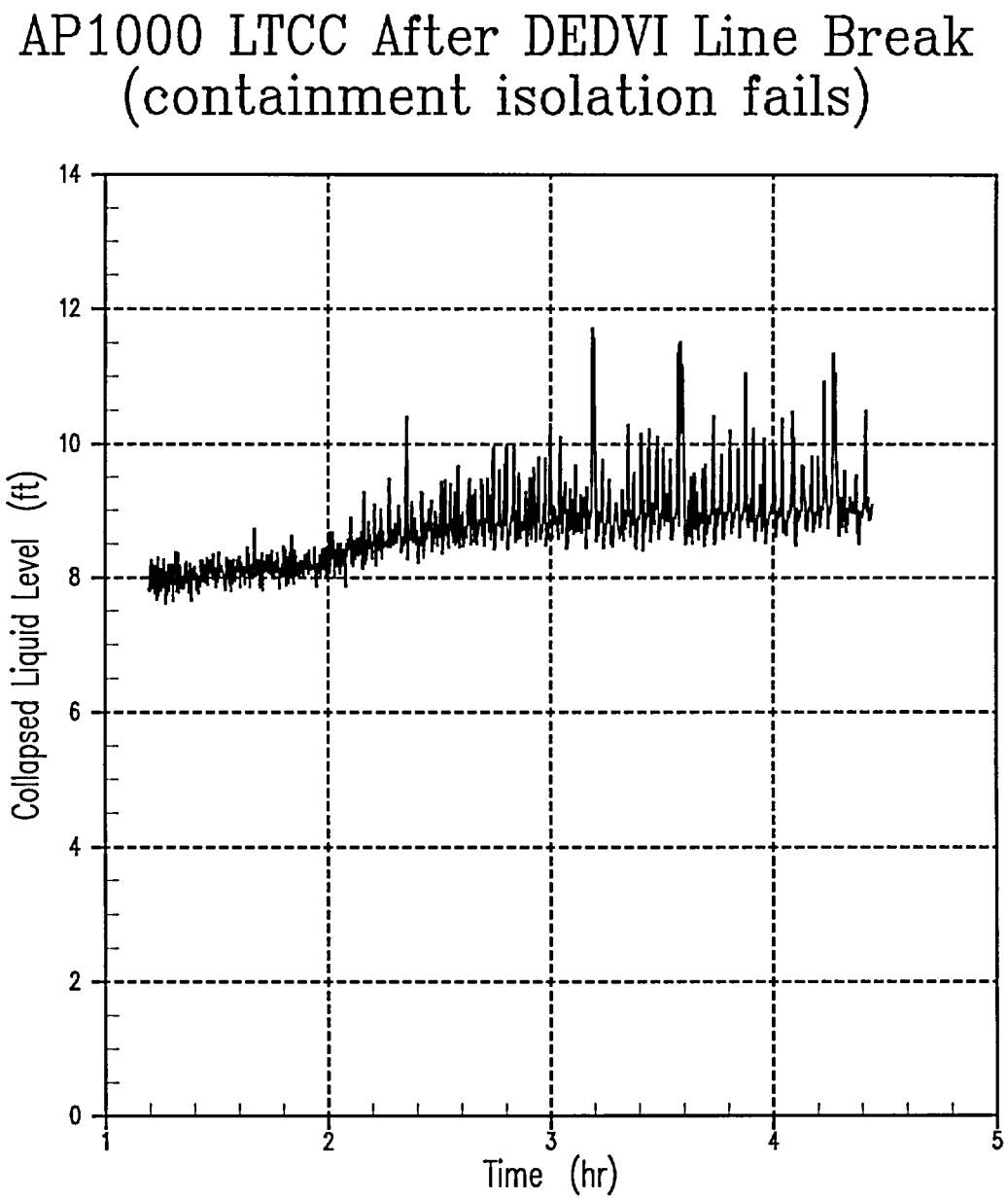


Figure RAI 720.013-16
Collapsed Level of Liquid Over the Heated Length of the Fuel (Case G)

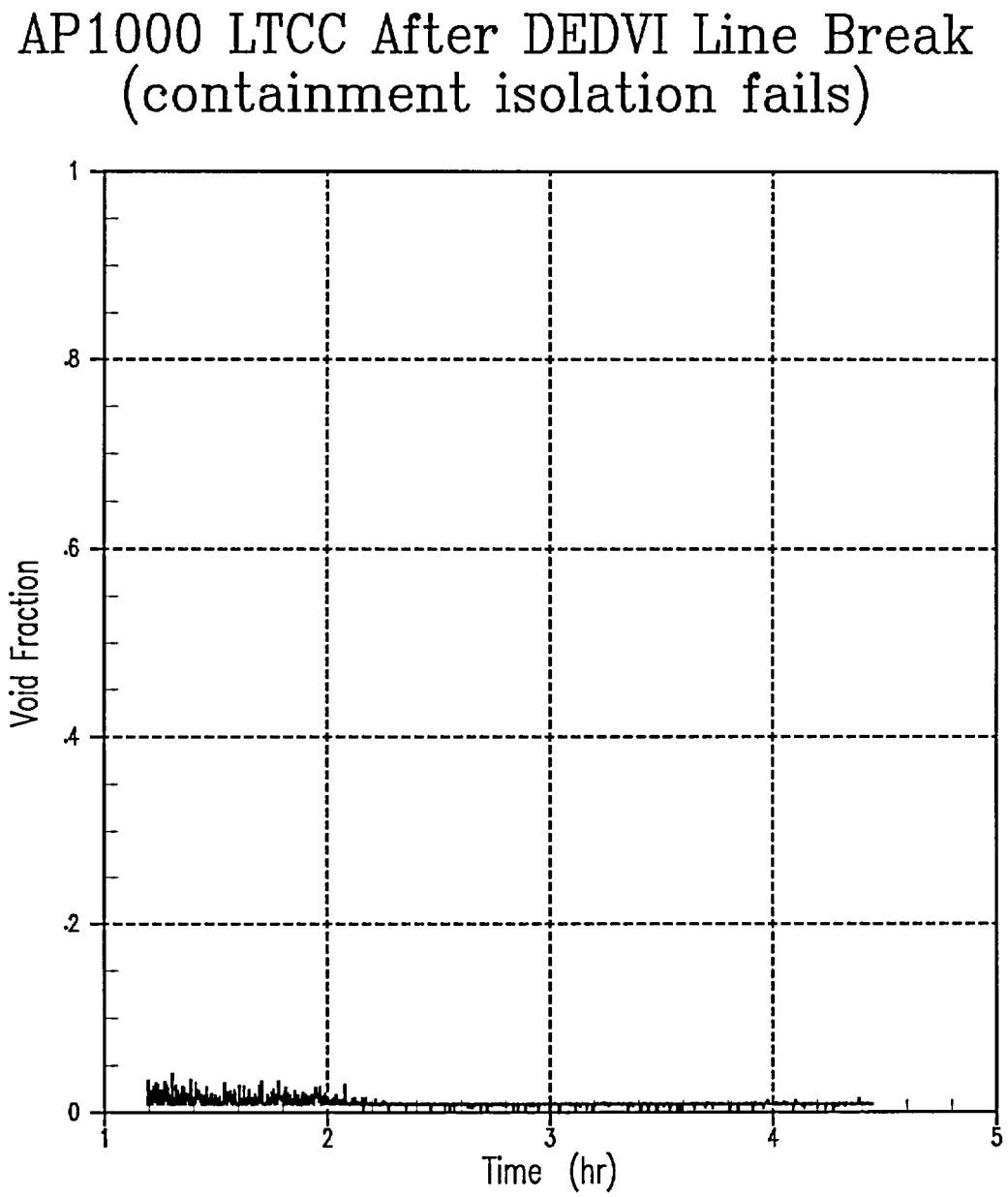


Figure RAI 720.013-17
Void Fraction in Core Cell Level 1 of 2 (Case G)

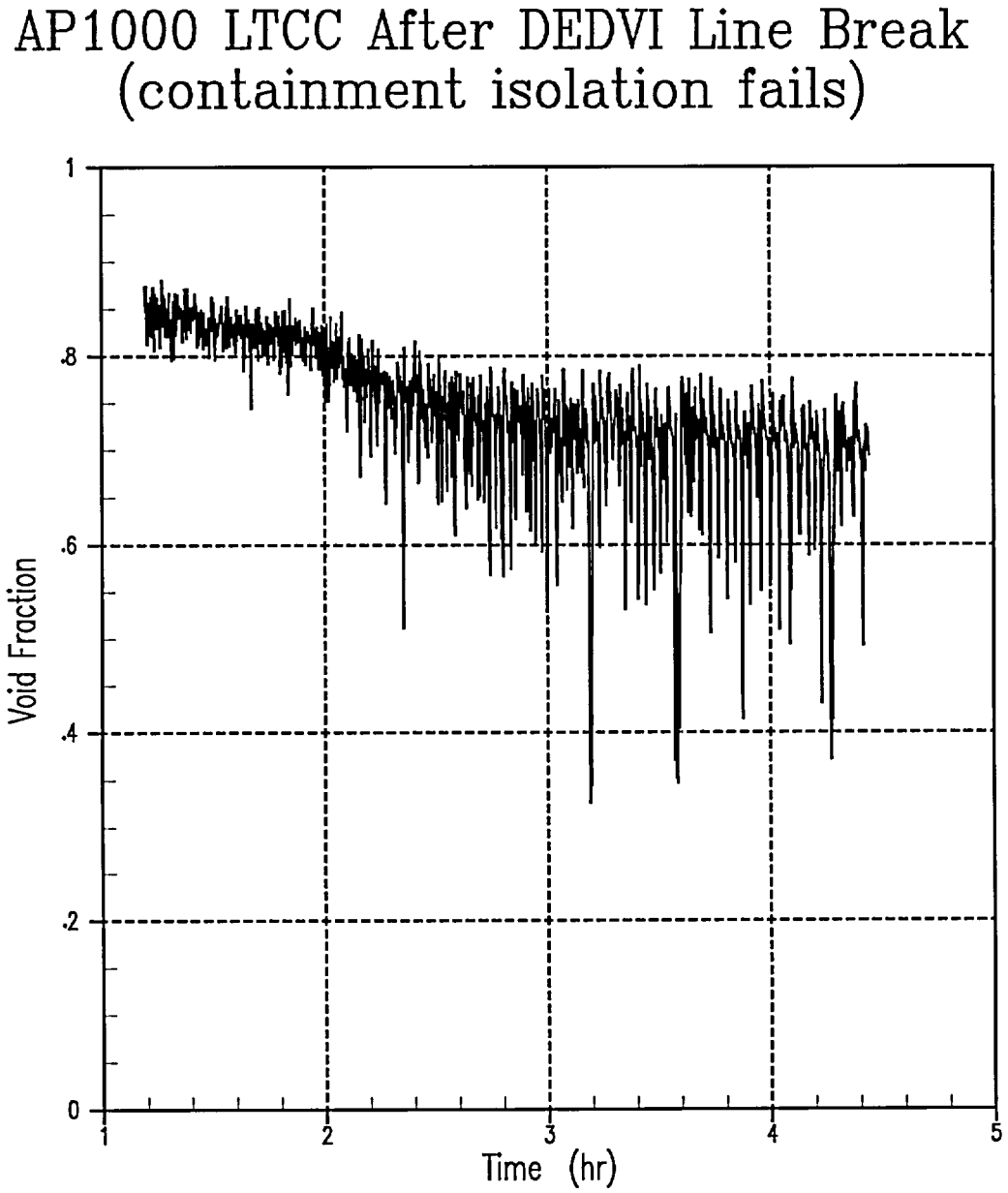


Figure RAI 720.013-18
Void Fraction in Core Cell Level 2 of 2 (Case G)

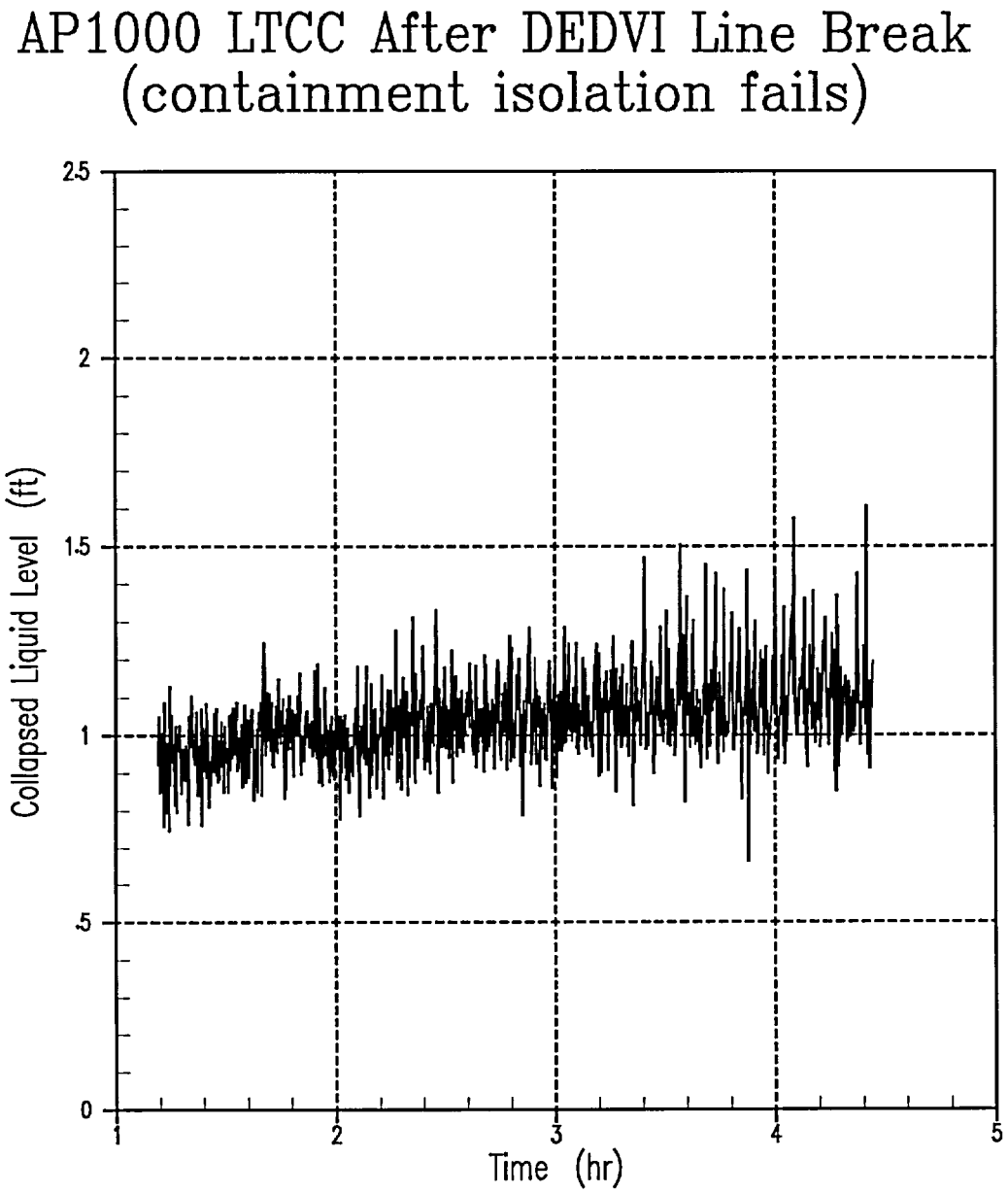


Figure RAI 720.013-19
Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (Case G)

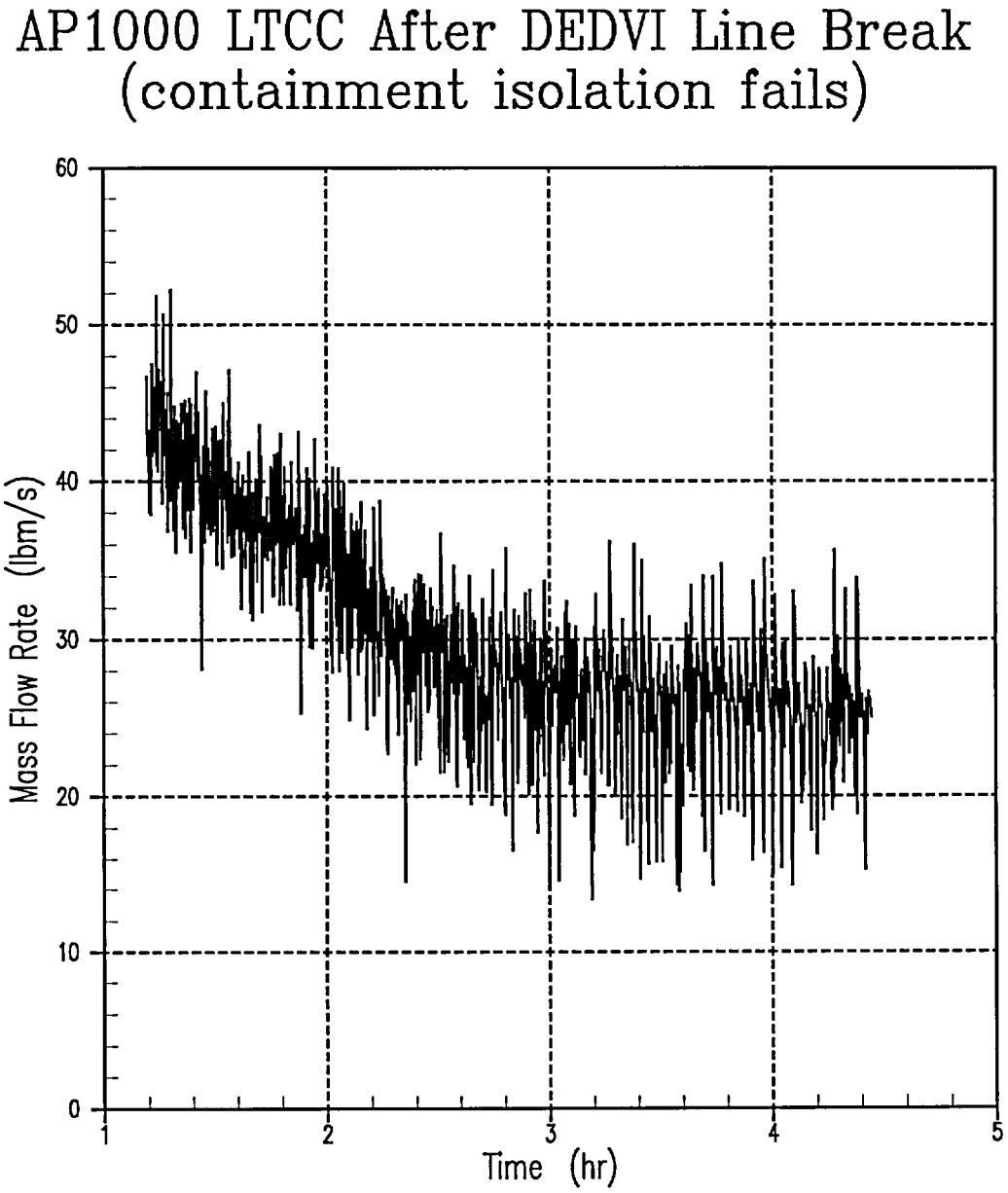


Figure RAI 720.013-20
Vapor Rate out of the Core (Case G)

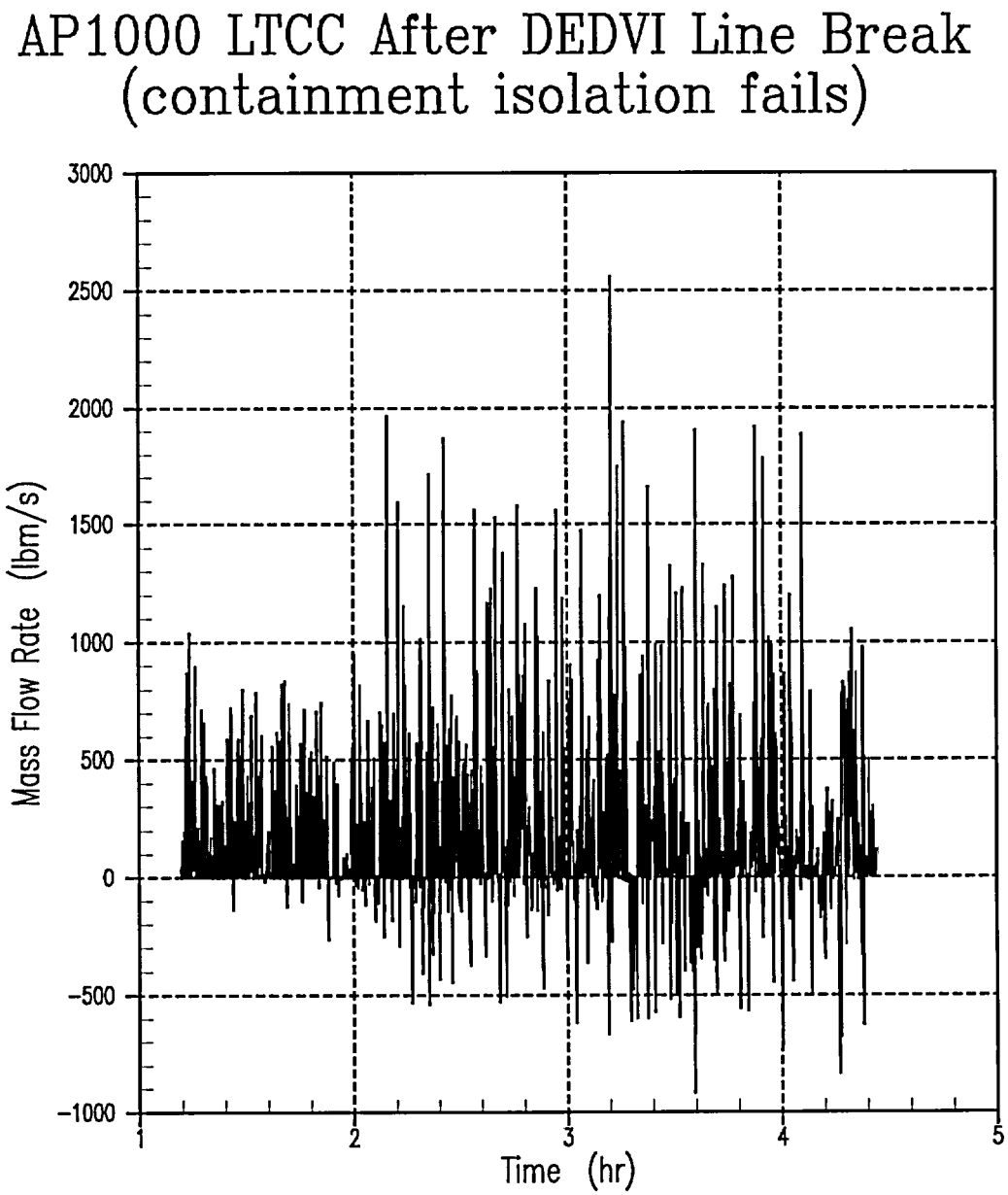


Figure RAI 720.013-21
Liquid Flow Rate Out of the Core (Case G)

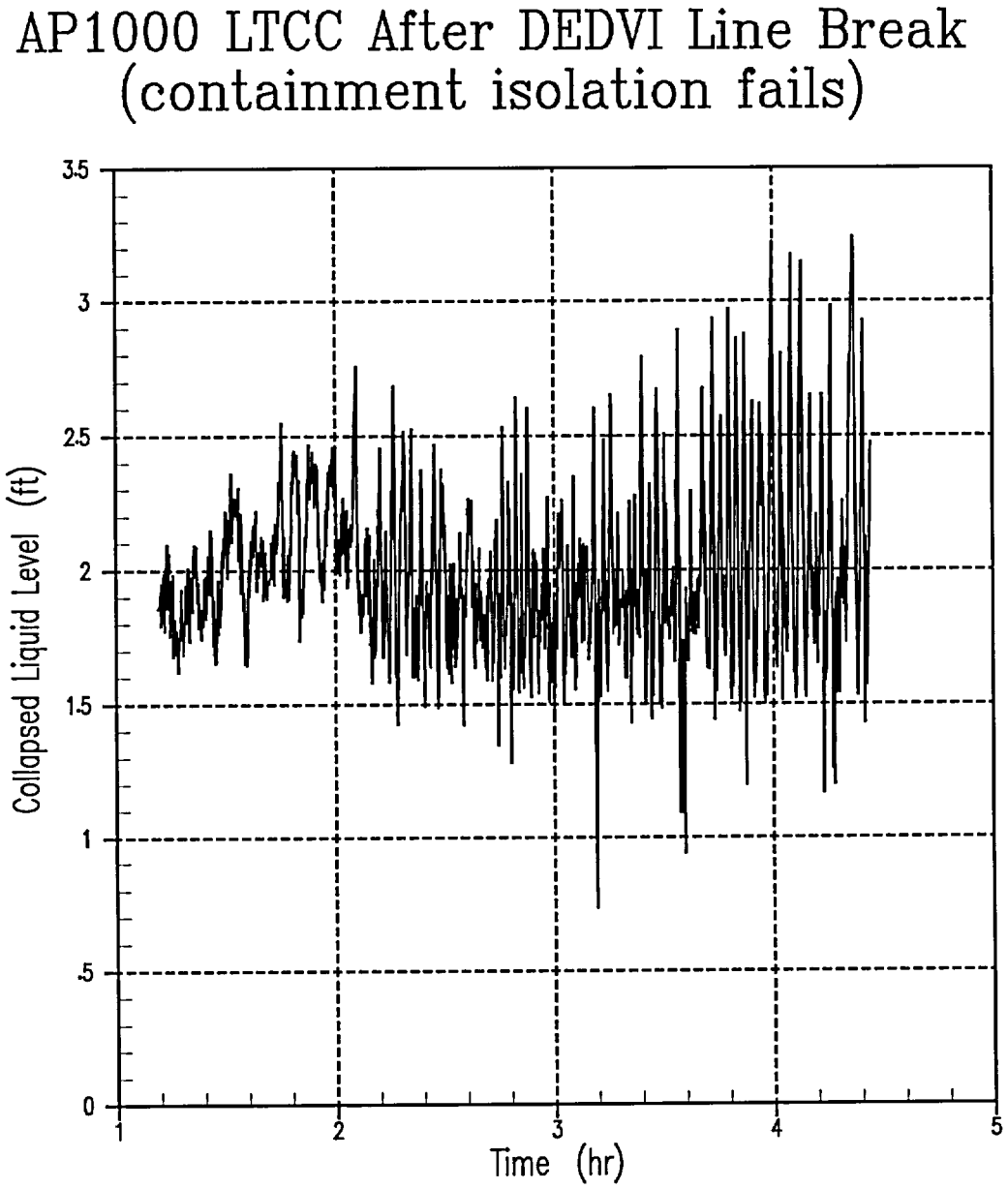


Figure RAI 720.013-22
Collapsed Liquid Level in the Upper Plenum (Case G)

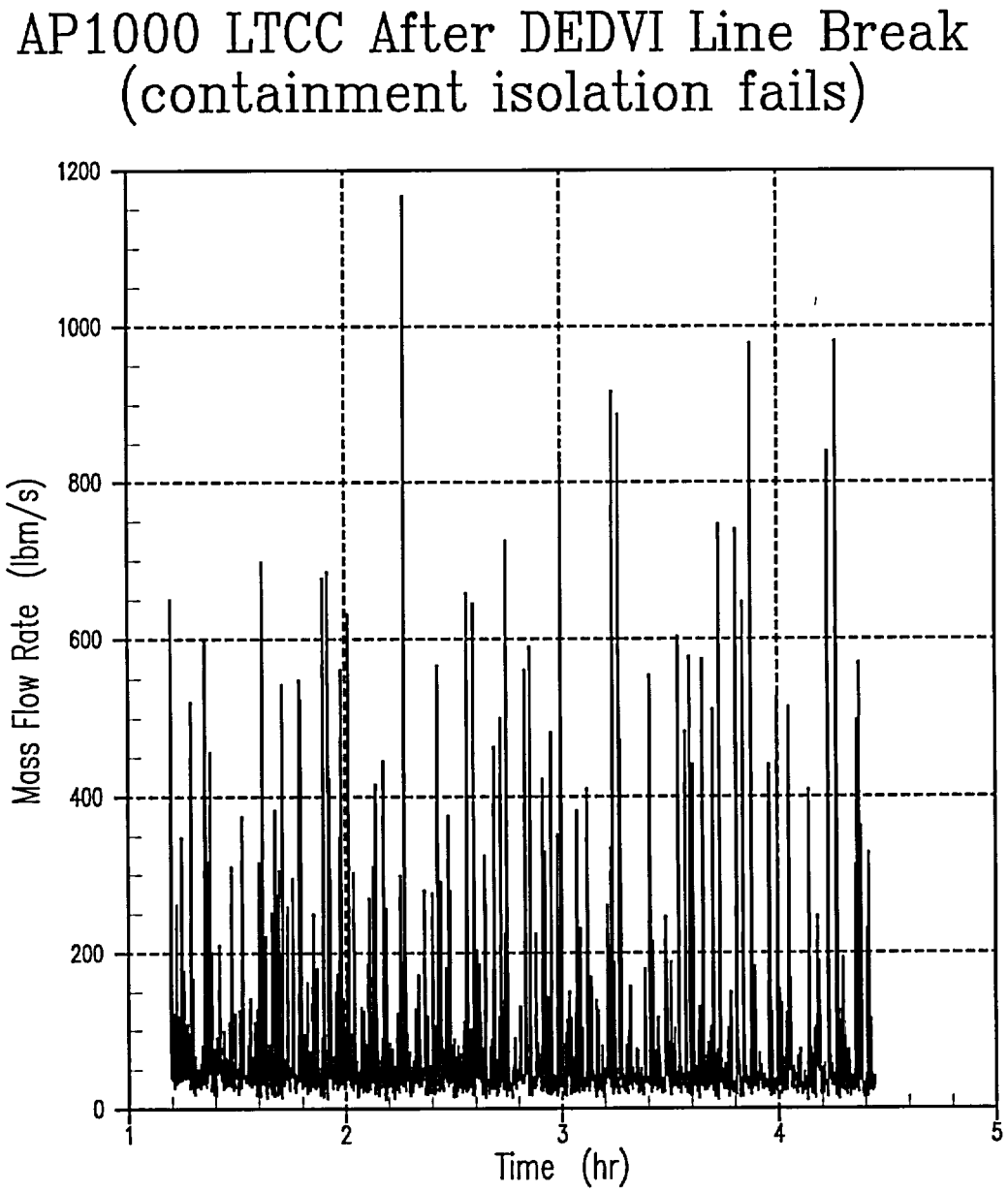


Figure RAI 720.013-23
Mixture Flowrate Through ADS Stage 4A Valves (Case G)

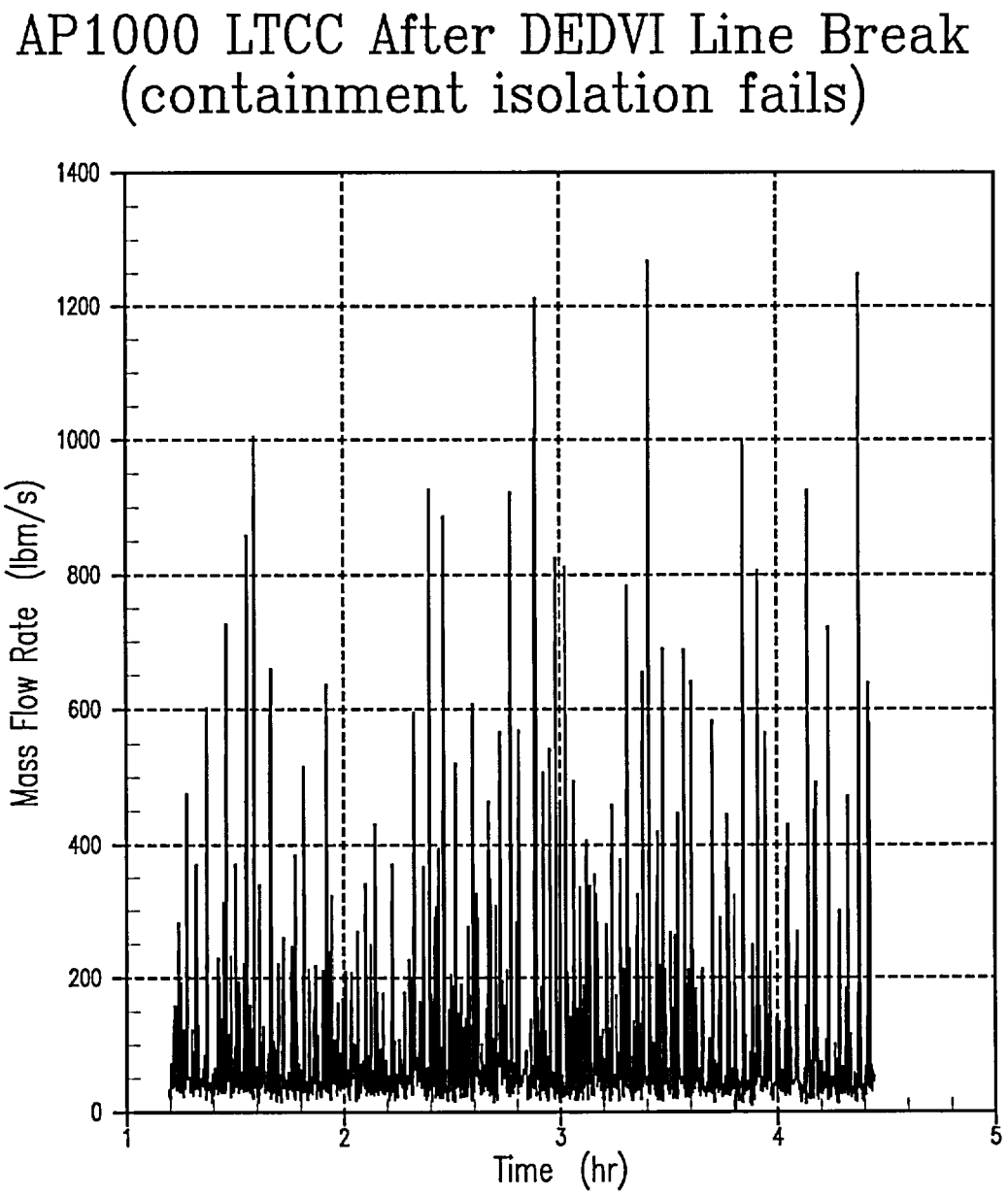


Figure RAI 720.013-24
Mixture Flowrate Through ADS Stage 4B Valves (Case G)

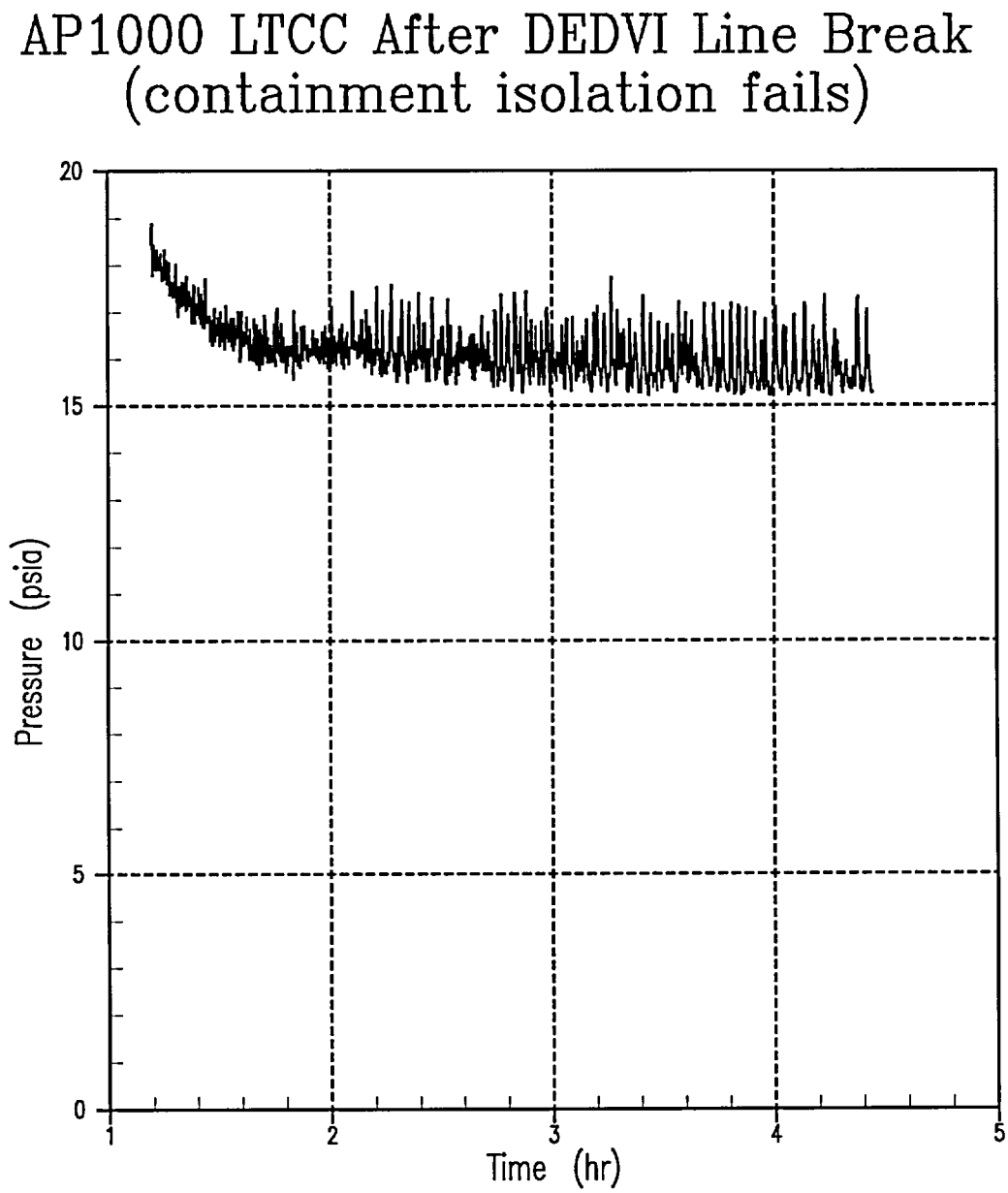


Figure RAI 720.013-25
Upper Plenum Pressure (Case G)

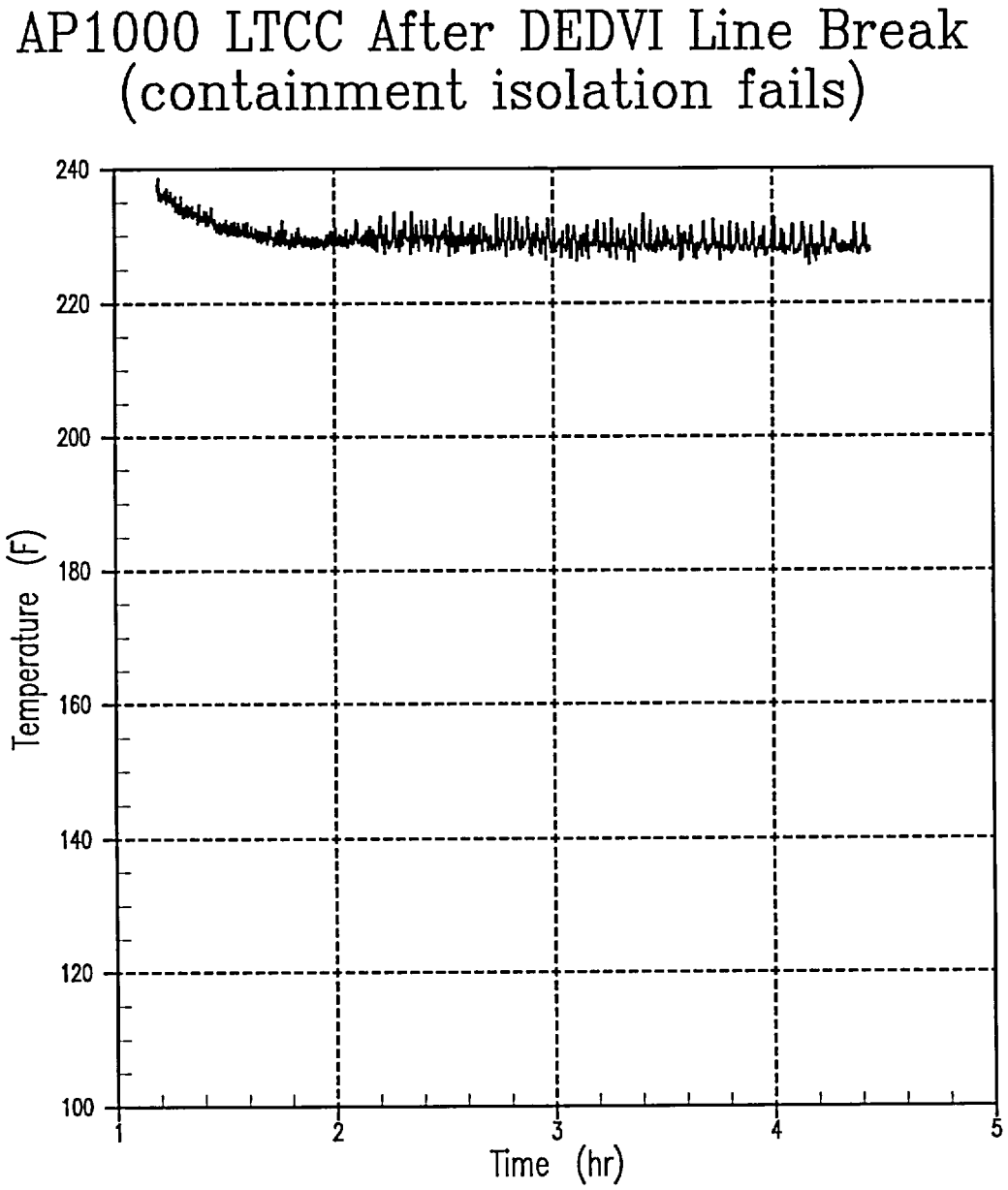


Figure RAI 720.013-26
PCT of the Hot Rod (Case G)

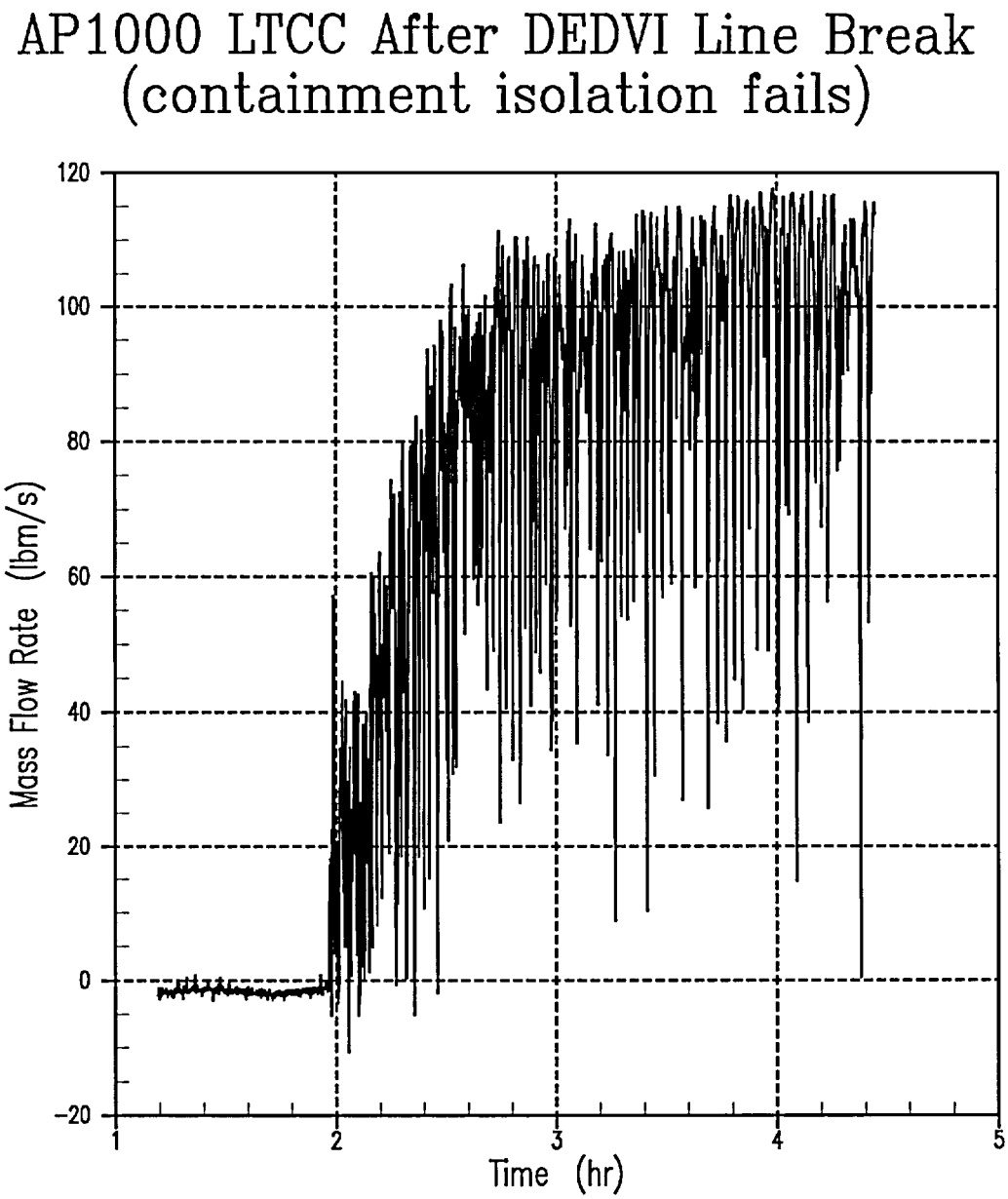


Figure RAI 720.013-27
DVI-A Mixture Flow Rate (Case G)

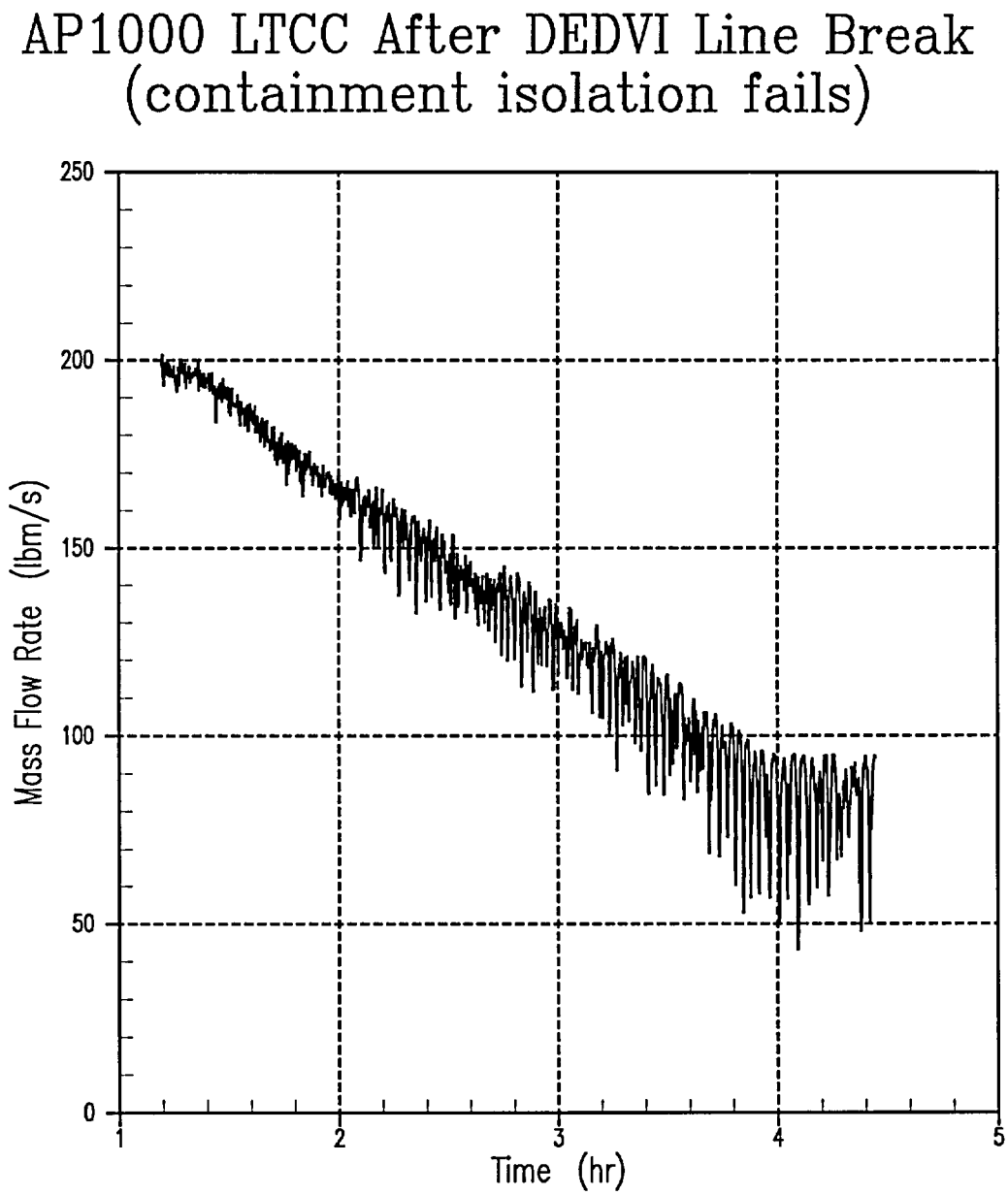


Figure RAI 720.013-28
DVI-B Mixture Flow Rate (Case G)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.021 (Response Revision 1)

Question:

Section 6.3.1.5, "Containment Isolation," states that "analyses (documented in Appendix A) were conducted to show that sufficient water for long-term cooling of the core will be retained in containment even if containment isolation is unsuccessful." A) Clarify where in Appendix A these analyses are documented.

Additional Question:

Open Issue: The response referenced an analysis using MAAP4 for which an 18 inch heating, ventilation, and air conditioning (HVAC) penetration was assumed to have failed opened. The NRC staff has not reviewed and approved the MAAP4 code for calculation of vapor and entrained droplet flow through a failed containment penetration. We require that the analysis be performed using a code that has been reviewed by the NRC staff. Assumptions for liquid and vapor flow from the failed containment penetration should be presented for staff review. The analysis was performed for a double ended DVI line which contains a 4 inch flow restrictor. Would sufficient water be retained within the containment for core cooling following larger break sizes?

Westinghouse Response: *<Note that the following is a revision to the Rev 0 response that addresses the additional question.>*

~~Long term cooling is discussed in Sections A2.2, A3.5, and A5.1 of Appendix A and explains the reasoning that AP600 long term cooling success criteria analyses can be used for AP1000. In addition,~~ **T**the response to RAI 720.013 contains long-term core cooling analysis for **two**~~a~~ sequence**s** with failure of containment isolation. **One case supports the PRA success criteria and the other case supports the PRA T&H uncertainty evaluation.**

The**se**~~is~~ core cooling analysis were~~as~~ performed using the WCOBRA-TRAC LTC model~~in support of the T&H uncertainty evaluation~~. The containment conditions for the**se**~~is~~ analysis were calculated outside WCOBRA-TRAC using the MAAP4 AP1000 model and were input to WCOBRA-TRAC. The WCOBRA-TRAC analysis shows that the core cooling is successful for **both of** the**se**~~is~~ events.

The **T&H uncertainty** sequence **that was** modeled in ~~both~~ MAAP4 and WCOBRA-TRAC is a DVI line LOCA with failure of containment isolation and other **risk important** failures. **Note that a DVI LOCA located in the PXS B room was assumed because it results in the largest**

containment flood volume and the lowest containment recirculation level. The available equipment includes:

- Four ADS stage 4 valves open and no ADS stage 1,2,3,
- Injection from one CMT **(the other CMT and both** ~~and one~~ accumulators **do not inject),**
- One IRWST injection line with one open valve
- Containment recirculation from one screen with two open valves,
- PCS water drain operating, and
- One containment penetration fails~~ure~~ to close.

Note that this case is an T&H uncertainty case, so it uses DCD methods and assumptions except for the additional failures. These assumptions include Appendix K decay heat and conservatively high line resistances. **The other LTC case supported the PRA success criteria and as a result used nominal analysis methods and assumptions including the limiting success criteria failures.**

An extreme containment isolation failure of an **sixteen**~~eighteen~~-inch HVAC penetration was assumed. **Note that no significant liquid carry over into the containment penetration is assumed because:**

- **The break is located in a PXS room below the 107' 2" floor level.**
- **The flow has a indirect pathway from this room up to the containment penetration which is located at about 125 feet elevation. In order for the flow to reach the penetration is must flow:**
  - **Around the pipes, valves, supports in the PXS B room and up through a large vent / ladder area in the ceiling**
  - **Around the large 107' 2" floor area about 30 feet horizontally to the back side of loop compartment 2 wall**
  - **Up through grating at the 118' 6" floor elevation to the inlet to the closest containment HVAC pipe**
- **The 107' 2" floor area is a large volume that is 26 feet high. This area provides for low velocities and many obstructions (pipes, structural steel, etc) that will cause water that might initially be entrained with the steam to settle out.**
- **The blowdown through the break only lasts until the ADS stage 4 valves open at about 3500 sec. Note that this analysis assumes that the CMT connected to the faulted DVI line fails to inject, which delays the ADS 4 actuation about 3000 sec. After the ADS stage 4 valves open, the steam / water flow from the RCS is located in the loop compartments which have an even more torturous path to the failed containment penetration.**

Figure 720.021-1 shows the containment water pool elevations out to 18 hours, indicating that the containment flood level is sustained at ~ 107 foot elevation in the long term. This level provides the driving head for injection through the intact DVI line (elevation of DVI nozzle ~ 99 feet). The water level in containment is sustained in the long term by return of condensate from the passive containment cooling (PCS) of the steam from the RCS. Figures 720.021-2, 720.021-3, and 720.021-4 show the containment pressure, containment leak flow, and containment heat removal, respectively. **The containment leak rate decreases to zero when the containment reaches atmospheric pressure in about 10,000 sec (2.8 hr). Figure 720.021-3 provides the containment leak flow rate vs time through the open penetration.** In the long term the **containment**~~result is atmospheric~~ pressure **remains at atmospheric pressure** ~~in containment~~ with decay heat steam being condensed by PCS and essentially no flow through the **open** containment **penetration**~~leak~~.

In order to verify that a DVI LOCA is the most limiting case, a DE CL break was also analyzed. This case also assumed that a containment HVAC penetration failed to close. The peak containment pressure was higher, but it decreased faster. The containment leak was also terminated earlier, in about 4000 sec (1.1 hours) with the containment water level at 109.3 ft. This water level is several feet above the level calculated for the DVI case, so it is less limiting.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

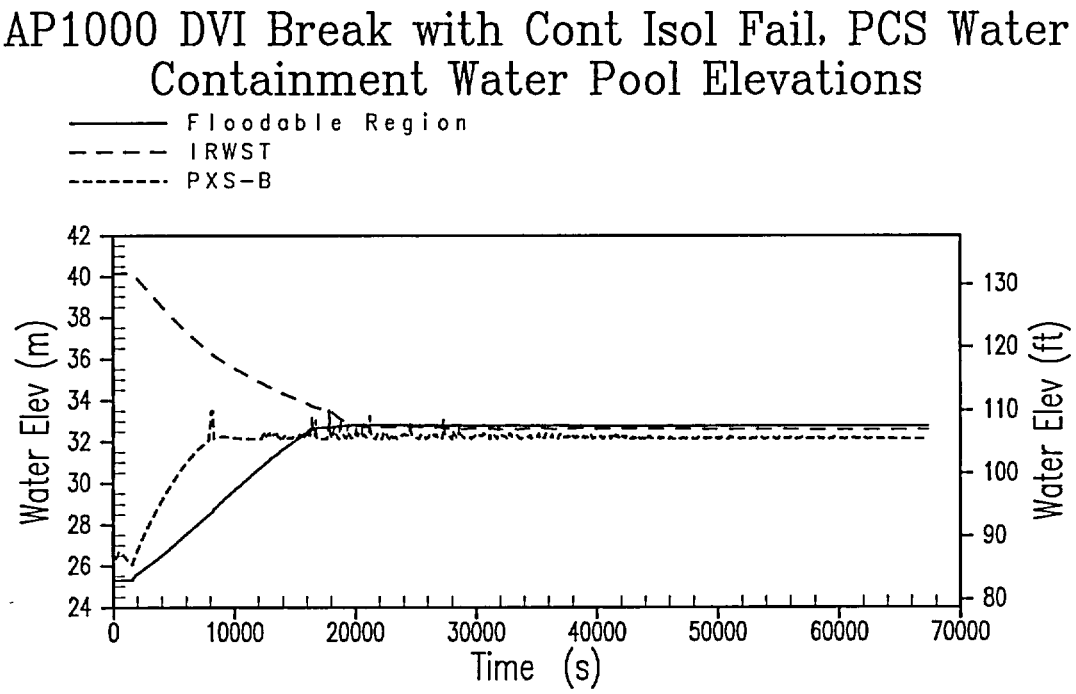


Figure 720.021-1 Containment Water Pool Elevations

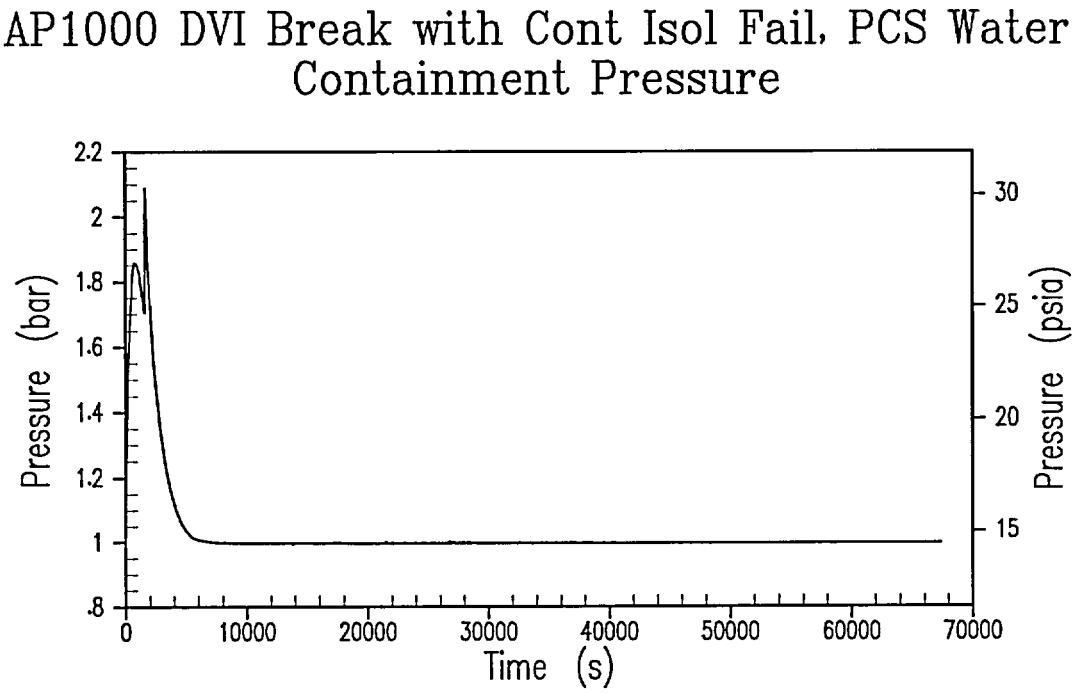


Figure 720.021-2 Containment Pressure

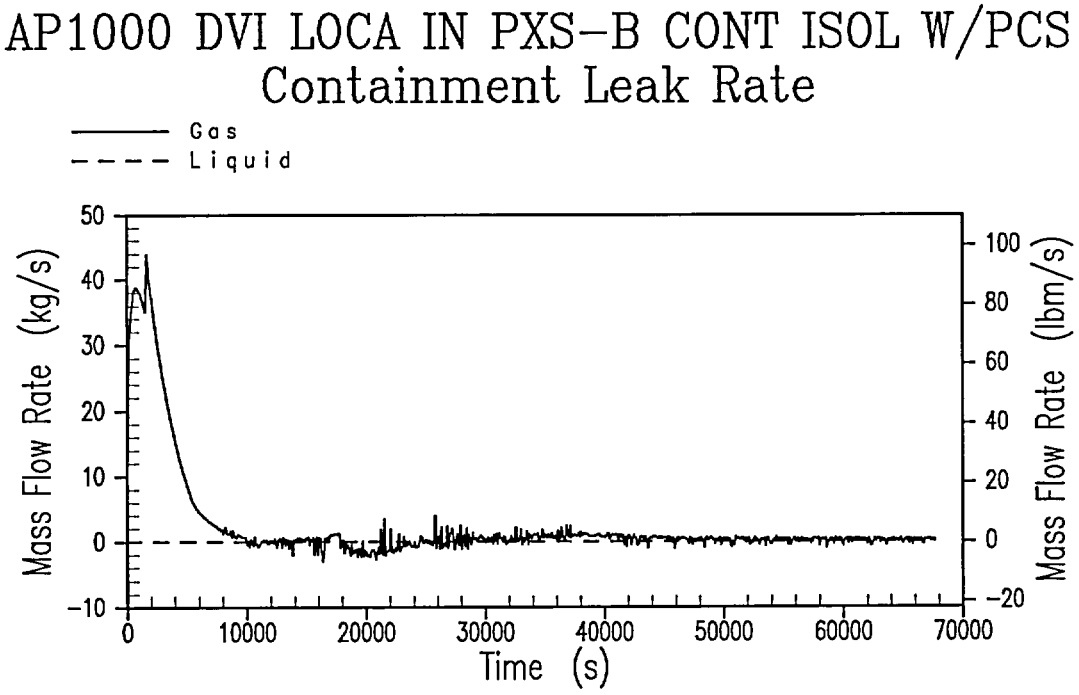


Figure 720.021-3 Containment Leak Flow Rate

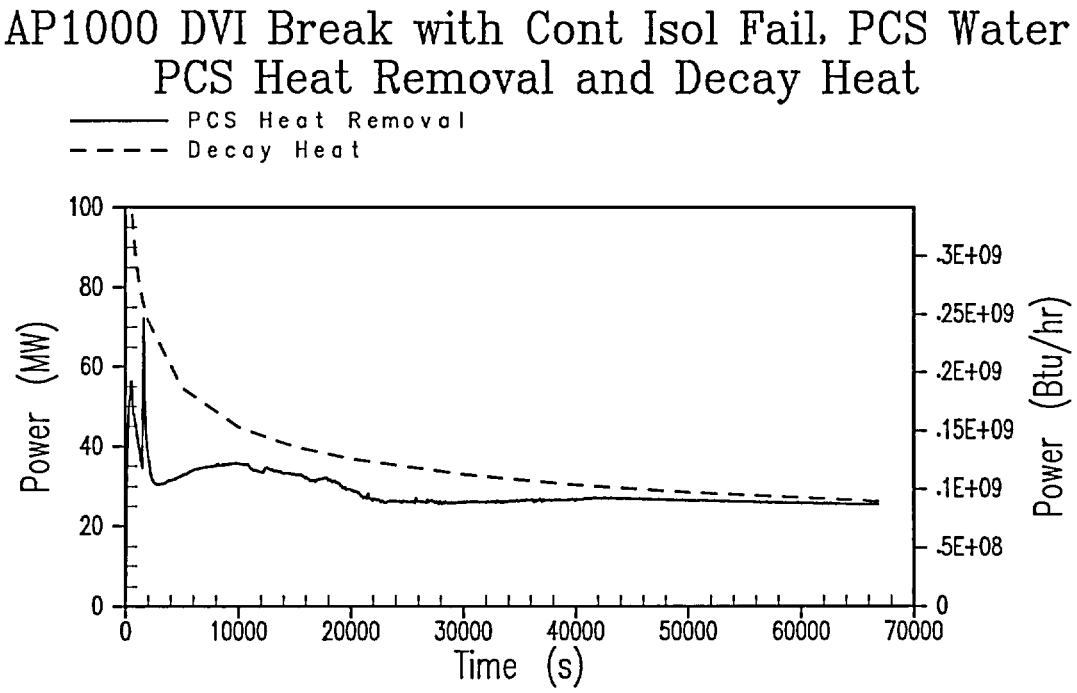


Figure 720.021-4 Containment PCS Heat Removal

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.025 (Response Revision 1)

Original Question:

Section 6.3.2 discusses key operator actions within the various accident sequences, and the available time for these operator actions. For example, Section 6.3.2.1 states that, for a loss of feedwater event, the maximum time available for manual PRHR actuation is determined to be greater than 45 minutes. Section 6.3.2.5 states that, for medium-break LOCA, the time available for operator action to actuate CMT injection is determined to be 10 minutes from the time the actuation signal occurs, and 20 minutes if accumulator injection is successful. With successful accumulator injection and PRHR operation, the available time for operator action to depressurize the RCS is determined to be approximately 20 minutes from the time CMT actuation occurs, and the time available to start RNS injection is determined to be 20 minutes. For small-break LOCA, steam generator tube rupture and transients, the time available to manually actuate CMT, and RCS depressurization is determined to be at least 30 minutes. The maximum time available for manual actuation of RNS is determine to be approximately 10 minutes from ADS actuation if PRHR has not actuated. In all cases, Appendix A is referenced for the determination of the maximum available time for the operator actions.

A. Clarify where in Appendix A these available operator action times are described.

B. Provide bases and determination of these available times for the AP1000 design.

NRC Additional Comments:

The table attached to the response states that at least 30 minutes of core cooling is available following a SLOCA, SG tube rupture (SGTR), or Transient with no CMT or accumulator injection. This conclusion is based on calculations done using MAAP for AP600 and MAAP calculations with 1 CMT for AP1000. Plant specific calculations should be provided for AP1000 since AP1000 has a higher power density than AP600. Uncertainty analyses using NOTRUMP should be provided.

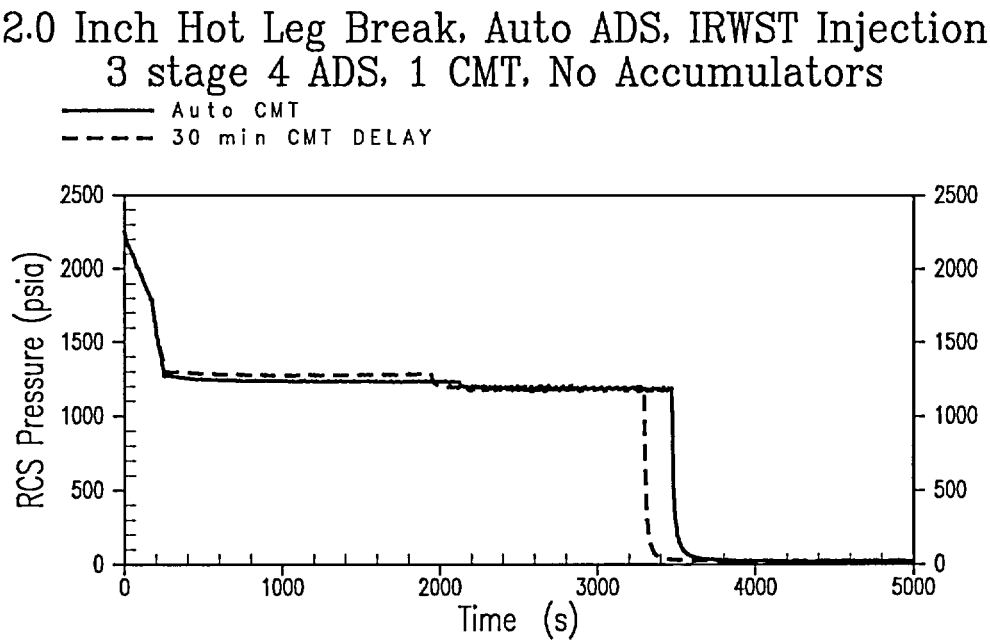
W to provide an analytical basis.

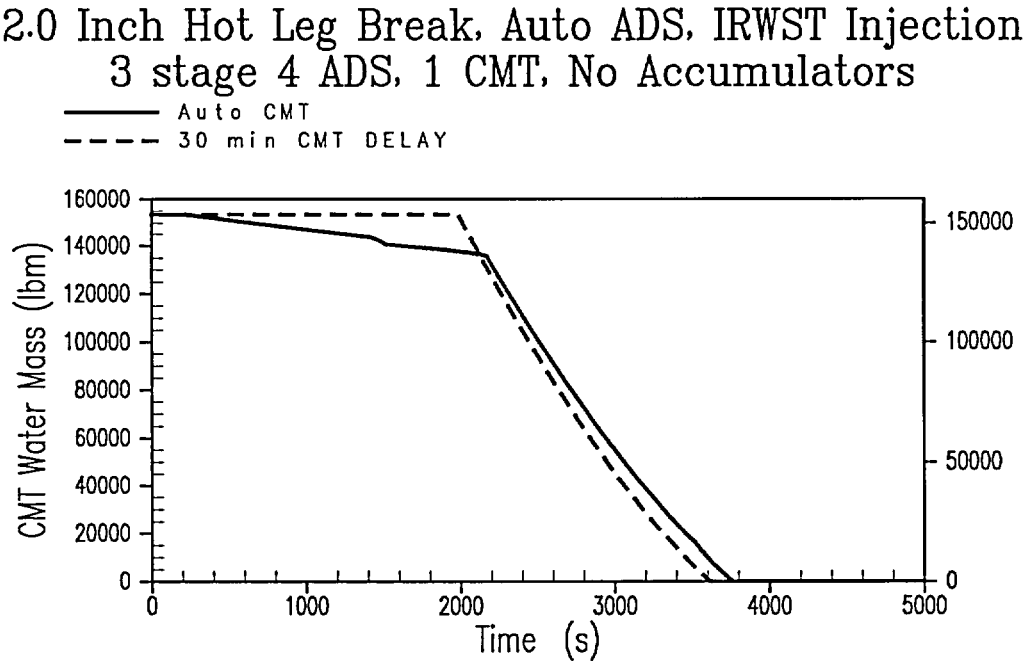
Westinghouse Response to Additional Questions: < The original response follows after this revised response. >

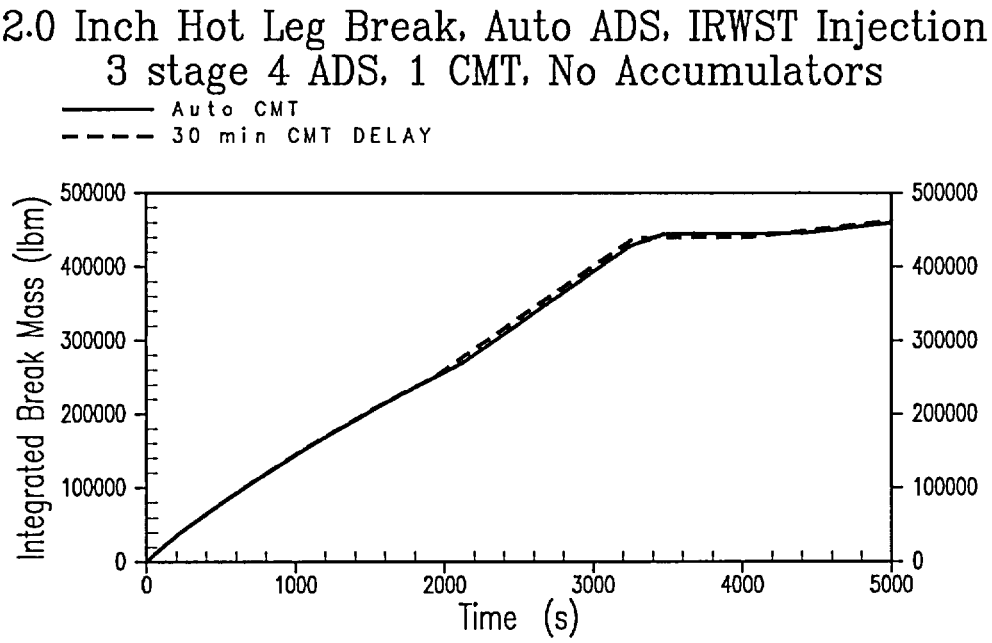
In the AP1000 PRA Appendix A, we show the results of MAAP analysis of a break at the upper end of the SLOCA break spectrum (2") and near the bottom end (0.5"). The results, of these two cases show that the larger break size is more limiting in terms of core uncovery. The 2" SLOCA has a core uncovery that starts at about 300 sec with a duration of ~ 400 sec and maximum depth of uncovery of ~ 3.5" The 0.5" SLOCA has a minor uncovery that starts much later (~ 11,700 sec), has a shorter duration (~ 100 sec) and has a smaller depth of uncovery (~ 2'). These results can be explained by the fact that decay heat is significantly less at the later times of ADS actuation seen in the 0.5" LOCA. The AP600 showed similar trends for SLOCAs and also for SGTRs and Transients with ADS actuation. As a result, the limiting event with respect to manual operator action to start the CMTs for these types of events is the 2" HL LOCA. From this information, it seems obvious that for Transients, SGTRs, and small SLOCAs that a delay in the actuation of the CMTs will not be significant with respect to the success of core cooling.

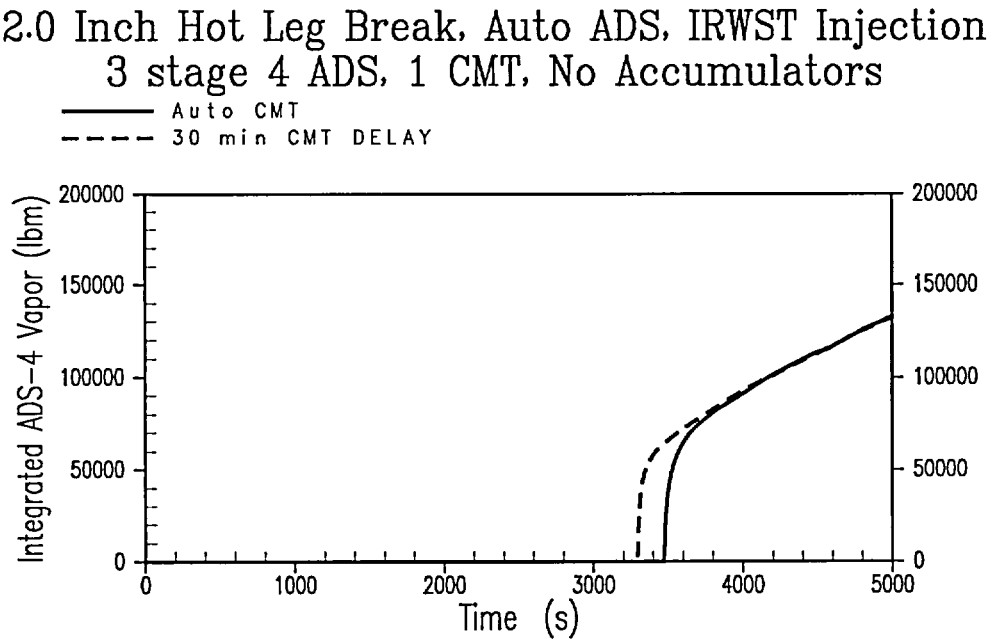
To address the most limiting case of a 2" HL SLOCA, we have performed an additional analysis. This analysis is the same as was performed for the Appendix A 2" HL LOCA with no accumulators and automatic CMTs, except that the CMTs were manually actuated 30 min after a CMT actuation signal was generated or about 2000 sec after the LOCA. As seen in the plot attached to this RAI, this analysis shows that the core does not uncover. This result justifies the 30 min. operator action time window.

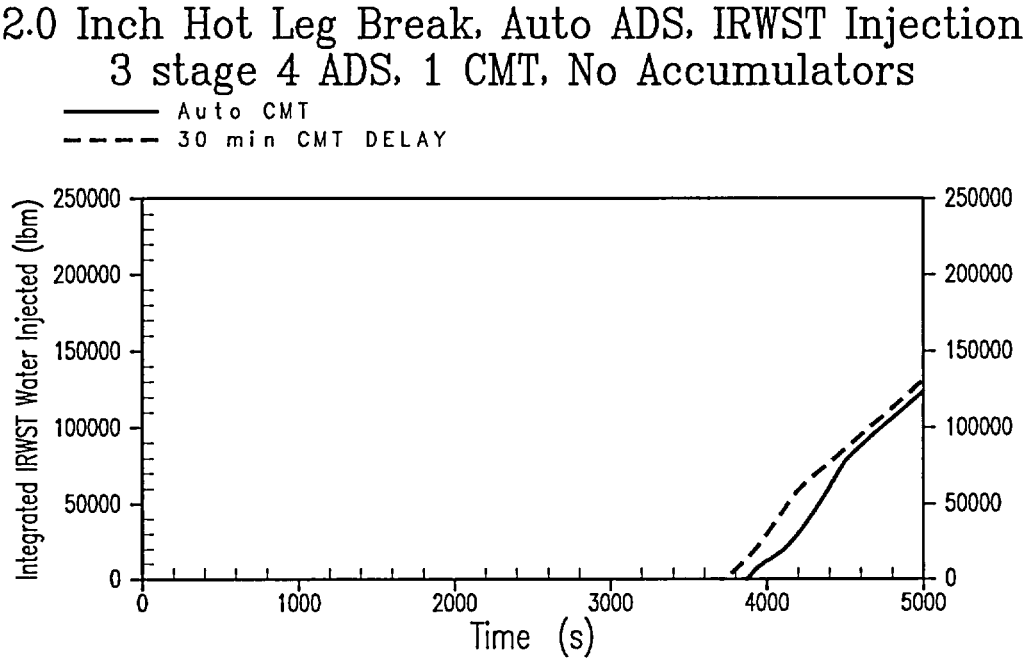
The T&H uncertainty analysis provided for AP1000 discusses how SLOCAs, SGTR and Transients are not limiting from a T&H margin and risk view point. This manual CMT actuation case, is less limiting than the automatic CMT actuation case since the core does not uncover. As a result, it is not necessary to perform NOTRUMP analysis to bound the T&H uncertainty of these manual CMT actuation cases.

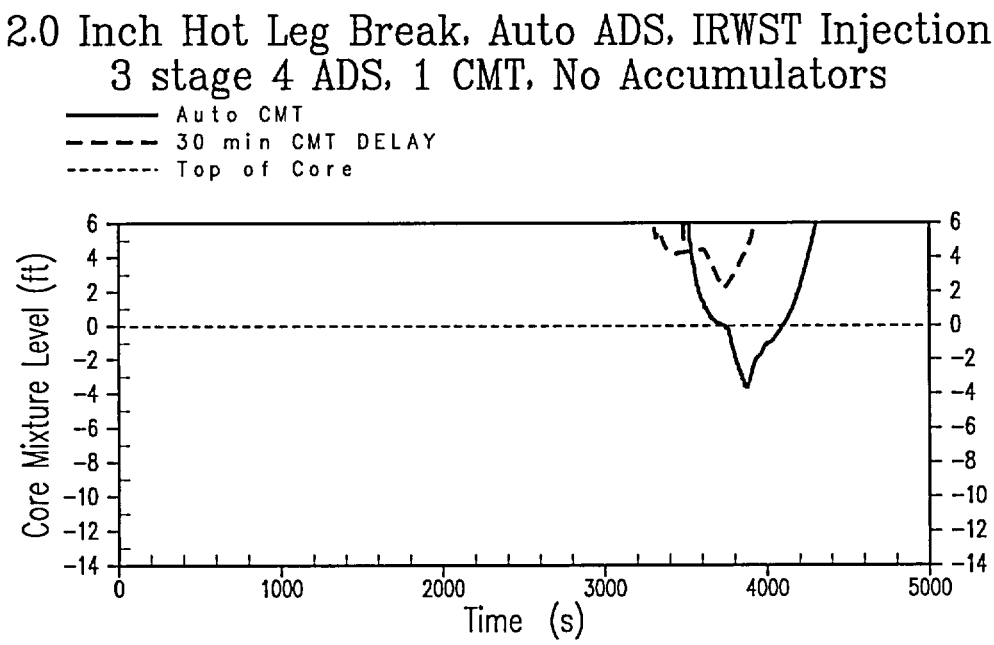

2.0 Inch Hot Leg Break, Auto ADS, IRWST Injection
3 stage 4 ADS, 1 CMT, No Accumulators
Auto CMT
30 min CMT DELAY
RCS Pressure (psia)
2500
2000
1500
1000
500
0
0
1000
2000
3000
4000
5000
Time (s)


2.0 Inch Hot Leg Break, Auto ADS, IRWST Injection
3 stage 4 ADS, 1 CMT, No Accumulators
Auto CMT
30 min CMT DELAY
CMT Water Mass (lbm)
160000
140000
120000
100000
80000
60000
40000
20000
0
150000
100000
50000
0
0
1000
2000
3000
4000
5000
Time (s)


2.0 Inch Hot Leg Break, Auto ADS, IRWST Injection
3 stage 4 ADS, 1 CMT, No Accumulators
Auto CMT
30 min CMT DELAY
Integrated Break Mass (lbm)
500000
400000
300000
200000
100000
0
0
1000
2000
3000
4000
5000
Time (s)


2.0 Inch Hot Leg Break, Auto ADS, IRWST Injection
3 stage 4 ADS, 1 CMT, No Accumulators
Auto CMT
30 min CMT DELAY
Integrated ADS-4 Vapor (lbm)
200000
150000
100000
50000
0
0
1000
2000
3000
4000
5000
Time (s)


2.0 Inch Hot Leg Break, Auto ADS, IRWST Injection
3 stage 4 ADS, 1 CMT, No Accumulators
Auto CMT
30 min CMT DELAY
Integrated IRWST Water Injected (lbm)
250000
200000
150000
100000
50000
0
0
1000
2000
3000
4000
5000
Time (s)


2.0 Inch Hot Leg Break, Auto ADS, IRWST Injection
3 stage 4 ADS, 1 CMT, No Accumulators
Auto CMT
30 min CMT DELAY
Top of Core
Core Mixture Level (ft)
6
4
2
0
-2
-4
-6
-8
-10
-12
-14
0
1000
2000
3000
4000
5000
Time (s)

Westinghouse Response:

The following table indicates the basis for the time windows cited in PRA Section 6.3.2:

EVENT	Time Window	Available Time	Basis
Loss of Feedwater	PRHR signal to SG dryout	45 min	AP600 PRA Appendix A Figures A-2 through A-5 show that SG initial inventory supports core cooling for more than 5000 sec. AP1000 SG inventory per unit of core power is higher than for AP600.
MLOCA	CMT signal to manual CMT with Accumulator	20 min	AP1000 PRA Appendix A Figures A3.3-2 through A3.3-19 show that one accumulator supports core cooling for the first twenty minutes.
MLOCA	CMT signal to manual CMT without Accumulator	Not credited	Text in AP1000 PRA Chapter 6 will be revised as shown in PRA revision included in this response.
MLOCA	CMT signal to manual ADS with Accumulator	20 min	AP1000 PRA Appendix A Figures A3.3-2 through A3.3-19 show that one accumulator supports core cooling for the first twenty minutes.
MLOCA	CMT signal to manual ADS without Accumulator	Not credited	Table 6.3 in AP1000 PRA Chapter 6 will be revised as shown in PRA revision included in this response.
MLOCA	Event initiation to manual RNS with Accumulator	20 min	AP1000 PRA Appendix A Figures A3.3-2 through A3.3-19 show that one accumulator supports core cooling for the first twenty minutes.
MLOCA	Event initiation to manual RNS without Accumulator	Not credited	Table 6.3 in AP1000 PRA Chapter 6 will be revised as shown in PRA revision included in this response.
SLOCA, SGTR, TRAN	CMT signal to manual CMT/ADS	30 min	AP600 PRA Appendix A Figures A-17 through A-19 show adequate core cooling for more than 5000 sec with no CMT or Accumulator injection. AP1000 PRA Appendix A Figures A3.2-2 through A3.2-7 show that AP1000 response for SLOCA timing is very similar to AP600.
SLOCA, SGTR, TRAN	ADS actuation to manual RNS with CMT without PRHR	10 min	AP1000 PRA Appendix A Figures A3.2-2 through A3.2-7 show that RNS injection at 10 minutes after ADS would support core cooling in the same manner as IRWST injection.

Several inconsistencies in AP1000 PRA Chapter 6 are corrected in the PRA revision shown below.

Design Control Document (DCD) Revision:

None

PRA Revision:

6.3.2.1 Time to Respond to Loss of Decay Heat Removal

For events involving a loss of main and startup feedwater to the steam generators, the PRHR system would be expected to remove decay heat. If PRHR failed to automatically actuate, the operators would be expected to manually actuate this function. If this were unsuccessful, the operators could initiate RCS depressurization using the ADS, in order to actuate core makeup tank or accumulator injection (i.e., feed and bleed).

The limiting loss of decay heat removal events are those that result in a loss of secondary side heat sink. A loss of main feedwater without startup feedwater (resulting from either station blackout or failure of startup feedwater (SFW)) is analyzed, since this results in the minimum secondary side inventory at the time of reactor trip. Of interest is the time at which steam generator heat transfer is significantly degraded.

Given a loss of main feedwater, with a subsequent failure of startup feedwater, the maximum time available for manual PRHR actuation is the time between generation of the PRHR actuation signal (assumed to occur on low steam generator wide range level), and the time that steam generator dryout occurs. This is determined to be greater than 45 minutes ~~(see Appendix A)~~. The Human Reliability Analysis (HRA) as documented in Chapter 30, assumes only a 30-minute time window for this action.

Once dryout occurs, following a loss of main feedwater with a subsequent failure of startup feedwater and PRHR, the operators must initiate depressurization via the ADS in about 30 minutes.

6.3.2.5 Time to Actuate Core Makeup Tanks and Depressurize RCS

The core makeup tanks are expected to automatically actuate and inject following a LOCA, or following a transient with loss of decay heat removal and subsequent relief through the pressurizer safety valves. As the CMTs inject and their level drops, the ADS is automatically actuated to depressurize the RCS to allow for accumulator injection (self actuated) and RNS operation (manually actuated) or gravity injection from the IRWST (automatically actuated).

If the CMTs fail to actuate automatically, the operator can manually actuate them. If this fails, the operator is directed to manually initiate RCS depressurization with the ADS and initiate RNS injection. Actuation of the ADS also opens the IRWST injection squib valves. Thus, if RNS subsequently fails, depressurization will continue to the point at which gravity injection from the IRWST will occur; in this case the operator would be required to actuate the IRWST squib valves as well.

<u>Large LOCA</u>

For a large LOCA in the PRA, the RCS depressurizes rapidly and the accumulators empty in several minutes. As a result, the HRA assumes that there is insufficient time to credit operator action to actuate IRWST injection or ADS valves. Although not required to provide core cooling, CMT injection is required to provide automatic IRWST and ADS actuation signals.

Medium LOCA

The time available for operator action to actuate CMT injection is defined as the time from the occurrence of the first signal that would alert the operators to the fact that CMT actuation should have occurred until the latest time at which CMT injection can begin such that core damage will be prevented without accumulator injection (assuming one train of RNS operates in injection mode after CMT injection). For medium LOCA, ~~this is determined to be 10 minutes from the time the actuation signal occurs (Appendix A), the CMT actuation signal occurs shortly after event initiation~~ **no credit is taken for this action if no accumulator is available.** The available time is 20 minutes if accumulator injection is successful.

For medium LOCAs, the time available for operator action to depressurize the RCS is defined as the time from event initiation until the time at which the ADS valves must begin to open in order to ensure that gravity injection can occur in time to prevent core damage (given that RNS has failed and the IRWST valves have opened). For cases with successful accumulator injection and PRHR operation, this is determined to be approximately 20 minutes from the time CMT actuation occurs (Appendix A), conservatively evaluated for breaks at the large end of the medium LOCA spectrum. For cases without accumulator injection or PRHR operation, operator action is not modeled in the PRA because of the shorter time available.

If the IRWST injection line valves fail to open automatically, the operators must actuate them in order to allow gravity injection. This action is actually the same action as ADS actuation, since the same manual control initiates both ADS and IRWST valves. As a result, the HRA considers appropriate system dependencies with respect to this action.

The maximum time available for operator action to start RNS injection is defined as the time from event initiation until the time at which one train of RNS injection is required to prevent core damage and prevent initiation of gravity injection (assuming one accumulator has injected). For cases with successful accumulator injection and PRHR operation, this is determined to be 20 minutes (Appendix A), conservatively evaluated for breaks at the large end of the medium LOCA spectrum. For cases without accumulator injection or PRHR operation, operator action is not modeled in the PRA because of the shorter time available.

Small LOCA, Steam Generator Tube Rupture, and Transients

For small LOCAs, steam generator tube rupture with failure to isolate the ruptured steam generator and equalize primary and secondary side pressures, and transients with loss of decay heat removal, the operators manually actuate CMT injection, and RCS depressurization, if the automatic actuations fail. The time available to take these actions is defined as the time from generation of an automatic signal to actuate CMTs until the time at which the ADS valves must begin to open in order to ensure that either RNS operation or gravity injection can occur in time to prevent core damage. Under this assumption, manual RCS depressurization is assumed to occur at the same time as manual CMT actuation. This is determined to be at least 30 minutes (Appendix A) for these types of initiating events.

If the IRWST injection line valves fail to open automatically, the operators must actuate them in order to allow gravity injection. This action is actually the same action as ADS actuation, since the same manual control initiates both ADS and IRWST valves. As a result, the HRA considers appropriate system dependencies with respect to this action.

Actuation of RNS is assumed to be performed in conjunction with actuation of ADS. The maximum time available for operator action to start RNS injection is defined as the time from ADS actuation until the time at which one train of RNS injection is required to prevent core damage and prevent initiation of gravity injection (assuming at least one CMT has injected). This is determined to be approximately 10 minutes (Appendix A, for small LOCA) if PRHR has not actuated. This corresponds to over an hour from event initiation. With PRHR operation, there would be several hours available.

Table 6-3 (Sheet 2 of 4)

SUMMARY OF SUCCESS CRITERIA FOR OPERATOR ACTIONS AND MISSION TIMES

Operator Action Identifier	Operator Action Description/Where Performed	Used in Event Tree Cases	Available Operator Response Time (Minutes)	Reference/ Basis
CIC-MAN01	Isolate containment following core damage/In control room	CIC	120 (except LLOCA pre-core damage)	O, E
CMN-MAN01	Actuate CMTs if automatic actuation fails/In control room	CM1A, CM2AB, CM2L, CM2P, CM2NL, CM2SL	(Timing consistent with associated recognition action - LPM-MAN01/2)	A O, E
CMN-REC01	Actuate CMT injection after core damage/In control room	CMBOTH, CM2LLT	>60 (Based on CMN-MAN01)	O, E
CVN-MAN00	Align CVS auxiliary spray for SGTR/In control room	CSAX	30	O, E
CVN-MAN02	Align CVS boration following ATWS/In control room	CSBOR1	60	O, E
CVN-MAN03	Start CVS standby pump if aligned pump fails/In control room	CVS1, CSP	30	A, E
DUMP-MAN01	Control steam dump during SGTR/ In control room	COND1	>30	A, E
FWN-MAN02	Start startup feedwater pumps if automatic actuation fails/In control room	SFW, SFW1, SFWM, SFWT	>30	O, E
FWN-MAN03	Start startup feedwater pumps if automatic actuation fails (LOOP)/ In control room	SFWP	30	O, E
HPM-MAN01	Recognize need for high pressure decay heat removal/In control room	PRL, PRP, PRS, PRT, SFW, SFW1, SFWM, SFWP, SFWT	30	A, O, E
LPM-MAN01	Recognize need for RCS depressurization/In control room	AD1, AD1A, ADA, ADR, ADS, ADT, ADZ, ADV, ADW, CM2AB, CM2SL	30 (Transients, SLOCA)	A, E
LPM-MAN02	Recognize need for RCS depressurization/In control room	ADQ, ADUM, ADAB, ADAL, ADB, ADM, ADR, ADRA, ADU, CM1A, CM2L, CM2NL, CM2P	20 ~~(MLOCA with accumulator injection)~~ ~~10 (MLOCA, without accumulator injection)~~	A, E

Table 6-3 (Sheet 4 of 4)

SUMMARY OF SUCCESS CRITERIA FOR OPERATOR ACTIONS AND MISSION TIMES

Operator Action Identifier	Operator Action Description/Where Performed	Used in Event Tree Cases	Available Operator Response Time (Minutes)	Reference/ Basis
RHN-MAN01	Align RNS after depressurization/ In control room	RNP, RNR	10 ~~(MLOCA without accumulator injection > 20 (Other)~~	A, E
RMN-MAN06	Recognize need and throttle RNS pump discharge valve if two RNS pumps are running with only one recirculation path available/ In control room	RECIRB, RECIRC, RECIRP, RNP, RNR	>30	E
RTN-MAN01	Perform controlled shutdown of the reactor	SDMAN	>60	E, O
SGHL-MAN01	Recognize need and actuate steam generator overfill protection/ In control room	SGHL	>30	O, E
VLN-MAN01	Recognize need and actuate containment hydrogen control system/In control room	VLH	>120	E
ZON-MAN01	Recognize need and start standby diesel generator/In control room	DGEN	>30	0

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.039 (Response Revision 1)

Question:

An important objective of the AP600 PRA was to provide input to the design certification process regarding the need for regulatory oversight of certain non-safety-related systems. The process used to identify SSCs for regulatory oversight as well as the type and level of such oversight was called Regulatory Treatment of Non-Safety-Related Systems (RTNSS) in the AP600 certification. The end results of this process were the "availability controls" documented in Section 16.3 of the DCD. Please provide similar information for AP1000. This information should account for uncertainties in the AP1000 PRA so that it can be used by the staff to make similar conclusions, about the need for non-safety-system oversight, as were made for the AP600 design (e.g., as documented in the AP600 FSER Chapter 19.1.7 "PRA input to the RTNSS Process.)"

Additional Question:

The staff requested Westinghouse to provide all important steps in the process of using PRA results to identify systems, structures and components (SSCs) for regulatory oversight as well as the type and level of such oversight for non-safety-related systems. This information should account for uncertainties in the AP1000 PRA so that it can be used by the staff to make similar conclusions, about the need for non-safety-system oversight, to those made for the AP600 design (e.g., as documented in the AP600 FSER Chapter 19.1.7 "PRA input to the RTNSS Process.)" Westinghouse did not provide this information with its response.

The following are additional comments on WCAP-15985 "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process".

1. Table 1.1 (P.1.6):

   The table lists non-safety systems evaluated in AP1000 RTNSS process in alphabetical order. It ends at steam generator. Are there other non-safety systems after steam generator evaluated with the RTNSS process?

2. Table 2-1 (p. 2-4):

   Table 2-1 lists only 5 non-safety systems and functions failed in PRA sensitivity studies, i.e., CVS, RNS, ECS, DAS and hydrogen ignitors. Does that mean other non-safety systems (such as the plant control system, component cooling water system, etc.) not on this list are assumed to be operational in the PRA sensitivity studies? If so, are they RTNSS important? Would they be subject to Technical Specifications or short-term

availability administrative control? If front-line systems are deemed important, then their supporting systems are also deemed RTNSS-important.

3. Section 2 states that the PRA sensitivity studies are based on the AP1000 baseline PRA, and that they include an evaluation of internal events that occur at-power. Are external events and low power/ shut-down internal events included in the sensitivity studies?

4. Section 2.2 states that the PRA CDF and LRF - with assumed failure of the nonsafety-related mitigation functions of the nonsafety-related SSCs - are reported in Chapter 50 of the AP1000 PRA report. However, Chapter 50 provides the importance and sensitivity analysis on CDF only. Clarify where in Chapter 50 describes the sensitivity studies for LRF?

5. Table 2-2 lists AP600/AP1000 PRA results for "baseline" and "without non-nuclear safety SSCs." The AP1000 shutdown, internal event, baseline PRA CDFs are listed as 1.2E-7 and 1.2E-8 in columns 2 and 3 for the shutdown internal events without and with manual DAS controls, respectively. Explain why the CDF for shutdown internal events without manual DAS controls is 10 times higher with manual DAS controls, whereas the LRF are the same for both cases.

6. Several places in Section 3, quote section 59 of the PRA as the source of various initiating event contributions to the LRF, e.g., Sections 3.2 and 3.4 states both the RCS leak initiating and main steamline break event contributes only 1.5% to the LRF, Section 3.5 lists LRF for various events, etc. Clarify where in Chapter 59 these LRFs for various events are provided.

7. Section 6 describes Post-72 hour Actions. Is the design of onsite equipment needed for post-72-hour support actions consistent with GDC 2 requirement for protection against natural phenomena? Where is this documented?

8. Section 10.2.2 describes the missions of various nonsafety-related plant systems, including the RNS for providing shutdown decay heat removal during RCS open shutdown conditions. Where is the RNS injection function to provide injecting cask loading pit water into the RCS following ADS actuation described?

9. Table 10-2, Investment Protection Short-term Availability Controls:

   A. 2.5 PCCWST and Spent Fuel Pool Makeup - Long-term Shutdown (P. 10-35):

   The "Operability" for Item 2.5 states that Long-term makeup to the PCCWST should be operable. Since the PCS recirculation pumps provide the capability to transfer water from the PCS ancillary water storage tank to the PCS water storage tank and spent fuel pool for post-72 hour support actions, why does the "Operability" requires only the

makeup to the PCCWST, but not SFP, to be operable? Clarify the basis for the PCS ancillary tank water volume of 725,000 gal in SR 2.5.1 (whether it is for 7 days or 3 days supply).

B. BASES for 3.3 AC power supplies - long-term shutdown (P. 10-56):

Clarify the basis for the ancillary fuel tank fuel volume of 600 gal in SR 3.3.1 (3 or 7 days supply?).

Westinghouse Response to Additional Question: < The following response is to the additional question; the original response is shown afterwards. >

WCAP-15985 "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process" provides a discussion of the PRA uncertainties in section 2.3. This discussion is the same as that provided for AP600. It addresses each identified PRA uncertainty and discusses nonsafety features that are able to compensate for them. The issues addressed include T&H uncertainty, equipment failure rates (IRWST check valves, squib valves, reactor coolant pump breakers), and importance of nonsafety equipment to PRA initiating event frequencies.

It is proposed that additional discussion be added to this section for each of these uncertainties and that the large LOCA initiating event frequency be added as another uncertainty. A revised version of this section is attached to this RAI response.

1. WCAP-15985, Table 1-1, is missing 7 systems. The following systems were inadvertently omitted and will be added to the end of this table:

   Storm Drainage
   Transmission Switchyard and Offsite Power
   Turbine Building Closed Cooling Water
   Turbine Building Ventilation
   Turbine Island Chemical Feed
   Turbine Island Vents, Drains and Relief
   Waste Water

2. Failing the front line nonsafety systems is a simple way of removing the mitigation capabilities of all the nonsafety systems from the AP1000 PRA. Once the front line systems have been removed the support systems have no capability of mitigating accidents. The Startup Feedwater System (SFWS) was also failed in this study. It was inadvertently omitted from Table 2-1; it will be added to this table in revision 1 of the WCAP.

Note that when a front line systems is captured as RTNSS important, then any support systems that are required to allow the front line systems to meet its RTNSS important mission are also considered RTNSS important.

3. The PRA sensitivity studies performed without nonsafety mitigating systems were not performed for external events because their contribution to CDF and LRF are small.

   The AP1000 PRA, revision 1, shows the following at power, core damage frequencies:

Internal Events	2.4E-7/yr
Internal Floods	8.8E-10/yr
Internal Fire	5.6E-8/yr

   From this information, it is concluded that it is not necessary for RTNSS to consider internal floods or fires.

   The AP1000 RTNSS evaluation did consider shutdown, internal events as shown in Table 2-1.

   A sentence will be added to section 2.0, second paragraph. The sentence will say, "Internal fire and flood events are not included in this evaluation because the AP1000 PRA quantification has shown them to have a much lower risk".

4. Chapter 50 of the PRA, revision 1, only includes CDF without nonsafety systems. Westinghouse has performed a sensitivity to determine the LRF with failure of the front line nonsafety features. This study will be added to the PRA chapter 50 in the next PRA revision.

5. As agreed with the staff, we will refer to the PRA sensitivity studies with credit removed for standby nonsafety systems as a focused PRA. The baseline PRA CDF and LRF should not change between columns 2 and 3 since the only difference is due to what nonsafety features are included in the Focused PRA. The CDF in column 2 is correct. The CDF in column 3 will be changed to 1.2E-7. The total CDF for column 3 is affected; it will change from 2.5E-7 to 3.6E-7.

6. The importance of initiating events to LRF was presented in the response to 720.057. This table will be added to the next revision of the PRA in Chapter 59.

7. The approach used for AP1000 is the same as was used for AP600, which is consistent with SECY-95-132. WCAP-15985, section 10.3.2, says that the long-term shutdown equipment should be available following seismic and high winds. This section also references DCD Table 3.2-3 and says that the supports for this equipment are Seismic II. DCD sections 1.9.5.3.1 (RTNSS) and 1.9.5.4 (Post-72 Hour Support Actions) provide additional information.

8. In section 10.2.2, the following will be added:

   "RNS (RCS intact)

   The RNS provides low pressure injection during accidents with ADS actuation."

9.A The OPERABILITY statement under availability control 2.5 will be changed as follows in both the WCAP and the DCD:

   "Long-term makeup to the PCCWST and the Spent Fuel Pool should be operable."

   The BASES for the capacity of the PCS ancillary tank will be clarified as follows in both the WCAP and the DCD:

   "The PCS recirculation pumps provide long-term shutdown support by transferring water from the PCS ancillary tank to the PCCWST and the spent fuel pool. The specified PCS ancillary water tank volume is sufficient ~~This water is used~~ to maintain PCS and spent fuel pool cooling during the 3- to 7-day time period following an accident. After 7 days, water brought in from offsite allows the PCCWST to continue to provide PCS cooling and makeup to the spent fuel pit. This PCCWST makeup function is important because it supports long-term shutdown operation. A minimum availability of 90 percent is assumed for this function during the MODES of applicability, considering both maintenance unavailability and failures to operate."

9.B The BASES for the availability control 3.3 will be changed as follows in both the WCAP and in the DCD,

   "The long-term AC power supply function involves the use of two ancillary diesel generators and an ancillary diesel generator fuel oil storage tank. The specified ancillary fuel oil storage tank volume is based on operation of both ancillary diesel-generators for four days. DCD subsection 8.3.1 contains additional information on the long-term AC power supply function."

Design Control Document (DCD) Revision:

The changes shown above in response 9A/B will be incorporated into DCD Table 16.3-2 in availability controls 2.5 and 3.3.

PRA Revision:

Add Table 50-26:

Table 50-26
CONTRIBUTION OF INIATING EVENTS TO LARGE RELEASE

	Initiating Event Category	% Contribution to LRF	LRF Contribution	Initiating Event Frequency
1	IEV-ATWS	17.11%	3.27E-09	4.81E-01
2	IEV-SGTR	15.87%	3.04E-09	3.88E-03
3	IEV-SPADS	13.14%	2.51E-09	5.40E-05
4	IEV-SI-LB	9.82%	1.88E-09	2.12E-04
5	IEV-TRANS	7.49%	1.43E-09	1.40E+00
6	IEV-SLOCA	5.94%	1.14E-09	5.00E-04
7	IEV-RV-RP	5.37%	1.03E-09	1.00E-08
8	IEV-MLOCA	4.71%	9.02E-10	4.36E-04
9	IEV-ATW-T	3.72%	7.12E-10	1.17E+00
10	IEV-LCOND	2.73%	5.22E-10	1.12E-01
11	IEV-LOSP	2.46%	4.70E-10	1.20E-01
12	IEV-LMFW	1.98%	3.80E-10	3.35E-01
13	IEV-LLOCA	1.65%	3.16E-10	5.00E-06
14	IEV-RCSLK	1.53%	2.93E-10	6.20E-03
15	IEV-SLB-V	1.22%	2.33E-10	2.39E-03
16	IEV-LMFW1	1.11%	2.12E-10	1.92E-01
17	IEV-CMTLB	1.03%	1.98E-10	9.31E-05
18	IEV-LCCW	0.72%	1.37E-10	1.44E-01
19	IEV-ATW-S	0.53%	1.01E-10	1.48E-02
20	IEV-LCAS	0.52%	1.00E-10	3.48E-02
21	IEV-POWEX	0.50%	9.49E-11	4.50E-03
22	IEV-PRSTR	0.45%	8.64E-11	1.34E-04
23	IEV-SLB-U	0.26%	4.97E-11	3.72E-04
24	IEV-LRCS	0.08%	1.58E-11	1.80E-02
25	IEV-SLB-D	0.05%	9.07E-12	5.96E-04
26	IEV-ISLOC	0.00%	4.74E-13	5.00E-11
	Totals	100.00%	1.91E-08	2.38E+00

PRA, Chapter 50.5.5:

Results

The calculations are performed on a personal computer using the SEN code for sensitivity analysis. The input file is taken from the AP1000 PRA CDF analysis. This file is CMTOT.OUT. The results of the sensitivity analysis are given in Tables 50-20 and 50-21.

Table 50-20 shows the contribution of the initiating events when no credit is taken for the above standby systems.

The output file contains 7269 cutsets. The top 50 of these cutsets are shown in Table 50-23.

This sensitivity analysis estimates that the CDF increases from 2.41E-07/year to 7.41E-06/year when no credit is taken for the standby systems CVS, SFS, RNS, DAS, and DGs.

A similar sensitivity study was performed to estimate the LF increase when no credit is taken for nonsafety standby systems. This study removed the hydrogen ignitors in addition to the CVS, SFS, RNS, DAS, and DGs. In this case, the LRF increases from 1.9E-8/year to 5.2E-6/year.

These results are limited by the way the sensitivity analysis is performed. Namely, if a CDF cutset does not appear in the CMTOT.OUT file due to cutoff probability, then it is resurrected in the present analysis.

Revision to WCAP-15985 "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process":

WCAP-15985, Table 1-1:

Table 1-1 (cont.)	Nonsafety-Related Systems Evaluated in AP1000 RTNSS Process
	Radiation Monitoring
	Radioactive Waste Drain
	Radiologically Controlled Area Ventilation
	Radwaste Building HVAC
	Raw Water
	Sanitary Drainage
	Secondary Sampling
	Security Lighting
	Seismic Monitoring
	Service Water
	Solid Radwaste
	Special Monitoring
	Special Process Heat Tracing
	Spent Fuel Pit Cooling
	Standby Diesel and Auxiliary Boiler Fuel Oil
	Stator Cooling
	Steam Generator
	Storm Drainage
	Transmission Switchyard and Offsite Power
	Turbine Building Closed Cooling Water
	Turbine Building Ventilation
	Turbine Island Chemical Feed
	Turbine Island Vents, Drains and Relief
	Waste Water

WCAP-15985, Section 2.0:

2 PROBABILISTIC RISK ASSESSMENT EVENT MITIGATION EVALUATION

PRA sensitivity studies were performed to quantify the importance of nonsafety-related systems in mitigating PRA events. These sensitivity studies, **refered to as the focused PRA,** calculate the CDF and LRF without reliance on nonsafety-related SSC mitigation. Nonsafety-related SSCs are considered not important for PRA mitigation if the resulting CDF is less than the NRC safety goal of 1×10^{-4} events per year and the resulting LRF is less than 1×10^{-6} per year. If nonsafety-related SSC mitigation functions are relied upon in these **focused** PRAs ~~sensitivity studies~~ to meet the safety goals, they will be assigned reliability/unavailability missions as appropriate and will be subject to additional regulatory oversight.

The **focused** PRA ~~sensitivity studies are~~**is** based on the AP1000 baseline PRA. They include an evaluation of internal events that occur at-power. **Internal fire and flood events are not included in this evaluation because the AP1000 PRA quantification has shown them to have a much lower risk.** Seismic margins are used to evaluate seismic events (section 9).

For the~~se~~ PRA sensitivity studies, the initiating event frequencies remain the same as in the baseline PRA. The mitigation functions of the nonsafety-related systems are failed, and then the CDF and LRF are calculated. If the CDF and LRF calculated in this **focused** PRA ~~sensitivity study~~ are acceptable and no mitigation credit is taken for nonsafety-related SSCs, then no additional regulatory oversight is necessary for the nonsafety-related SSCs.

Table 2-1 lists the AP1000 nonsafety-related systems modeled in the baseline PRA and assumed to fail to provide mitigation. Table 2-1 also contains a list of the safety-related systems modeled in the PRA.

2.1 EVALUATION

The **focused** PRA ~~sensitivity study~~ is performed using the same methodology as the baseline PRA. In the quantification of the CDF and LRF, the failure probability of each nonsafety SSC is set to 1. The **focused PRA** ~~sensitivity study~~ for CDF and LRF **is**~~are~~ described in Chapter 50 of the AP1000 PRA report. In addition, the improvement to these results, due to crediting manual Diverse Actuation System (DAS) controls, was estimated by evaluating the cutsets associated with these results.

2.2 RESULTS

The PRA CDF and LRF – with assumed failure of the nonsafety-related mitigation functions of the nonsafety-related SSCs – are reported in Chapter 50 of the AP1000 PRA report. Summary results from the sensitivity studies and a comparison with the AP1000 and AP600 baseline CDF and LRF are also found in Table 2-2.

Table 2-2 shows that use of the~~se~~ **focused PRA** ~~sensitivity studies~~ results in higher CDF/LRF than obtained in the AP600 focused PRA. This result is caused by not having all events lose offsite ac power

upon reactor trip. As a result, the Protection and Safety Monitoring System (PMS) is required to actuate passive safety features, such as the rods, PRHR heat exchanger (PRHR HX), and containment isolation. These **AP1000 focused PRA** ~~sensitivity studies~~ indicates that the LRF will be above the safety goal. By crediting the manual DAS controls, the LRF as well as the CDF are reduced so that the PRA safety goals are met.

As a result, the following DAS manual controls need to be credited for PRA mitigation:

- Reactor trip
- PRHR HX and in-containment refueling water storage tank (IRWST) gutter valves
- Core makeup tank (CMT) isolation valves
- Automatic Depressurization System (ADS) stages 1, 2, 3, and 4
- IRWST injection isolation valves
- Containment recirculation isolation valves
- PCS water drain valves
- Containment isolation valves

Since the DAS manual controls are credited to meet the LRF safety goal, it was concluded that these DAS manual controls should be included in the AP1000 Technical Specifications. (See section 10.4 for a draft of the DAS manual control Technical Specification.)

WCAP-15985, Table 2-1, first part:

Table 2-1 Systems and Functions Credited in Probabilistic Risk Assessment Sensitivity Studies
Nonsafety-Related Systems and Functions Failed in PRA Sensitivity Studies
Chemical and Volume Control System (CVS)
Normal Residual Heat Removal System (RNS)
Startup Feedwater System (SFWS)
Main AC Power System (ECS) (diesel only)
Diverse Actuation System (DAS)
Hydrogen ignitors

WCAP-15985, Table 2-2:

Table 2-2 AP600/AP1000 PRA Results for Baseline and Without Non-Nuclear Safety SSCs			
	AP1000[1]	AP1000[2]	AP600
CDF, At-Power, Internal			
Baseline	2.4 E-7	2.4 E-7	1.7 E-7
~~Without NNS SSCs~~	7.4 E-6	2.3 E-6	~~N/A~~
Focused	~~N/A~~	~~N/A~~	7.7 E-6
CDF, Shutdown, Internal			
Baseline	1.2 E-7	1.2 E-~~7~~8	1.0 E-7
~~Without NNS SSCs~~	(3)	(3)	~~N/A~~
Focused	~~N/A~~	~~N/A~~	4.1 E-7
CDF, Total Internal			
Baseline	3.6 E-7	3.6~~2.5~~ E-7	2.7 E-7
~~Without NNS SSCs~~	< 1 E-4 [3]	<< 1 E-4 [3]	~~N/A~~
Focused	~~N/A~~	~~N/A~~	8.1 E-6
Safety goal	1.0 E-4	1.0 E-4	1.0 E-4
LRF, At-Power, Internal			
Baseline	1.9 E-8	1.9 E-8	1.8 E-8
~~Without NNS SSCs~~	5.2 E-6	4.3 E-7	~~N/A~~
Focused	~~N/A~~	~~N/A~~	5.5 E-7
LRF, Shutdown, Internal			
Baseline	2.0 E-8	2.0 E-8	1.5 E-8
Without NNS SSCs	(4)	(5)	N/A
Focused	N/A	N/A	2.6 E-7
LRF, Total Internal			
Baseline	3.9 E-8	3.9 E-8	3.3 E-8
~~Without NNS SSCs~~	> 1 E-6 [4]	< 1 E-6 [5]	~~N/A~~
Focused	~~N/A~~	~~N/A~~	8.2 E-7
Safety goal	1.0 E-6	1.0 E-6	1.0 E-6

Notes:

1. Assumes NNS SSCs in Table 2-1 fail
2. Assumes NNS SSCs in Table 2-1 fail, except for manual DAS controls listed in section 2 2.
3. Based on AP600/AP1000 results, the AP1000 CDF during shutdowns without NNS SSCs is estimated to be an order of magnitude less than the CDF at-power without NNS SSCs.
4. The base AP1000 LRF at-power without NNS SSCs exceeds the safety goal without including the shutdown LRF.
5. Based on AP600/AP1000 results, the AP1000 LRF during shutdowns without NNS SSCs is estimated to be about the same as the LRF at-power without NNS SSCs

Definitions.

N/A = not applicable

NNS = non-nuclear safety

WCAP-15985, Section 10.2.2:

- RNS (RCS Open)

  The RNS provides shutdown decay heat removal during RCS open shutdown conditions.

- **RNS (RCS Intact)**

  The RNS provides low pressure injection during accidents with ADS actuation.

- CCS (RCS Open)

  The CCS provides cooling to support RNS shutdown decay heat removal operation during RCS open shutdown conditions.

WCAP-15985, Table 10-2, Availability Controls 2.5 and 3.3:

The word changes shown in response 9A/B will be implemented in these availability controls.

Westinghouse Response: < This is the original (unchanged) response. >

WCAP-15985 "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process" provides the basis for the implementation of the RTNSS policy statement. The AP1000 implementation of the RTNSS policy statement is consistent with the approach that was taken for the AP600. Note that PRA sensitivity studies with nonsafety systems failed were used to evaluate the importance of nonsafety features in PRA accident mitigation, instead of the focused PRA sensitivity study. The AP1000 PRA, revision 0, includes the CMF sensitivity study; the LRF sensitivity study was performed recently based on the response to RAI 720.057 and will be added to the PRA in revision 1.

Use of these sensitivity studies increases the need for passive safety feature actuation signals since non-safety AC power is assumed to be available after each accident except for loss of offsite power. As a result, some non-safety manual Diverse Actuation System (DAS) controls are required to meet the licensing PRA safety goals. These manual DAS controls are captured as RTNSS important and additional regulatory oversight is provided. Since these manual DAS controls meet the technical specification screening criteria for PRA importance, a Technical Specification is added on these manual DAS controls. The additional Technical Specification is included in the revised AP1000 Technical Specifications that are submitted as part of the response to RAI 630.001.

These DAS manual controls were the only additional non-safety featured captured in the AP1000 RTNSS evaluation. The list of non-safety features that have short-term availability controls is the same for AP1000 and AP600.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

See the attached Technical Specification for DAS that is included in the revised AP1000 Technical Specifications.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.043 (Response Revision 1)

Original Question:

Time windows available for operator actions in AP1000 are shorter than for AP600 (see Table 35-6 of the AP1000 PRA). For every human action in the Level 2 PRA, please describe the basis for the revised time estimates, and their impact on human error probabilities (HEPs) and containment performance (i.e., large release frequency).

Additional Question:

The RAI noted that time windows available for operator actions in AP1000 are shorter than for AP600 and requested that W provide an assessment of the shorter times on human error probabilities and containment performance. The response addressed these impacts for 1 operator action, but 3 additional actions have shorter times in AP1000 and were not addressed.

Revise response to update Table 43C-1 and modify HRA assumptions if necessary (LLOCA).

Westinghouse Revised Response:

The following is a revision of the original Westinghouse response. It addresses the additional NRC question above. The 3 top events that had shorter times in the original response are:

- **IS Manual containment isolation for an accident with core damage**
- **PC Manual initiation of PCS water drain**
- **IG Manual initiation of hydrogen ignitors**

The time windows for all three of these events have been increased as described in this response.

Attachment 43C at the end of the AP1000 PRA Chapter 43 provides an evaluation of operator actions pertinent to the containment event tree nodes. Attachment 43C includes a comparison to the AP600 time windows and HEPs, and a sensitivity study showing the impact on large release frequency if larger HEP is assumed for the shorter time window for cavity flooding. Some inconsistencies were noted between Attachment 43C and Table 35-6 in the AP1000 PRA and will be revised as indicated below.

In addition, the time windows for AP1000 were verified with AP1000 MAAP4.04 analyses. These MAAP4.04 analyses are discussed in AP1000 RAI response 720.042.

For top event DP, the following MAAP4.04 analyses were used to test the 30-minute time window:

- 1A-3,
- 1A-4,
- 1AP-3,
- 1AP-4,
- 6E-1, and
- 6L-1.

Of these runs, only 1AP-3 and 6L-1 exhibited core damage. Both of these runs support the 30-minute time window. The 3A event was not analyzed because it is an ATWS that does not have any operator actions associated with it.

For top event IS, the following MAAP4.04 analyses were used to test the 350-minute time window:

- 1AP-3,
- 3BE-2,
- 3BE-4,
- 3BE-5,
- 3BE-6,
- 3BE-8,
- 3BE-9,
- 3BL-1,
- 3BL-2,
- 3D-1,
- 3D-2,
- 3D-3, and
- 3D-5.

1A-3, 1A-4, 1AP-4 and 3BE-1 were not used because they do not exhibit core damage. All of these runs support the 350-minute time window, **except for 3D-1. Note that the 50 minute Tw is completely consistent with the HEP listed in Table 43C-1 since it is greater than the operator action time (30 min) and provides more than the minimum time (5 minutes) for recovery actions. Sequence 3D-1 is a spurious opening of one ADS stage 4 valve with injection from 2 accumulators. No injection comes from the CMTs or the IRWST. Core damage starts at 26 minutes, which is slightly less than the 30 minute operator action time used to calculate the HEP for CIC-MAN01. In addition, it would not allow for recovery actions (requires an additional 5 minutes). As a result, for this sequence the HEP would be expected to be somewhat higher.**

We have evaluated the importance of CIS-MAN01 compared to automatic containment isolation. If the operator action of cic-man01 is set to failure in the containment isolation fault tree, the change in the containment isolation failure probability is very small (1.65E-03 to 1.658E-03). This is understandable because each containment penetration only has simple redundancy of isolation valves and it has very reliable automatic isolation from both the PMS and DAS.

In summary, the use of the HEP shown in PRA Table 43C-1 is considered reasonable because:

- **All of the dominant core melt sequences in RAI 720.042 have a Tw greater than 50 minutes except for one (3D-1).**
- **Sequence 3D-1 has a Tw of 26 minutes, which will allow for good operator action reliability, although somewhat less than that listed in Table 43C-1.**
- **Even if this operator action is assumed to have a failure rate of 1, the impact on containment isolation reliability is negligible.**

For top event IR, two IVR analyses (see AP1000 PRA Chapter 39) were used to verify the 5-minute time window.

For top event PC, AP1000 containment **air-only cooling**~~venting~~ analyses **(AP1000 PRA Chapter 40)** were used to verify the **18 hour** ~~60-minute~~ operator action time window. **Figure 40-5 shows the AP1000 containment pressure vs time with air-only cooling for both nominal and bounding analysis. At approximately time 0 the containment pressure exceeds the actuation setpoint for the PCS water drain valves. The containment pressure where the probability of containment failure is 1 E –3 is about 85 psig (AP1000 PRA Chapter 42, Table 42-3). The AP1000 would reach this pressure in about 18 hours with bounding assumptions (Figure 40-5). As a result, at least 18 hours would be available for the operators to actuate the PCS water drain valves, which supports the 18 hour time window.** ~~The time between containment pressure of 3 bars and containment pressure of 6 bars (containment pressure corresponding to a 10^{-3} probability of failure from AP1000 PRA Chapter 42) supports the 60-minute time window.~~

For top event VNT, the same venting analyses were used to verify the 60-minute operator action time window. Venting was assumed to start at ~5.5 bars (containment pressure with 5% failure probability from AP1000 PRA Chapter 42, less 10 psig). The time window is the time from ~5.5 bars to 6 bars (containment pressure corresponding to a 10^{-3} probability of failure from AP1000 PRA Chapter 42). The 60-minute time window is supported by these analyses.

For top event IG, AP1000 hydrogen mixing analysis 3C-1 was used to verify the 15~~0~~-minute operator action time window. The time from core-exit temperature > 1200 °F to the global flammability hydrogen limit (0.10 mass fraction hydrogen) is 19 minutes per the analysis. This supports the 1~~0~~5-minute operator action time window.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

Table 35-6

SUMMARY OF OPERATOR ACTIONS CREDITED ON CONTAINMENT EVENT TREE

Top Event	Description of Operator Error	Event ID	Cue(s)	Time Window
DP	Failure to recognize need for post-core-uncovery RCS depress during small LOCA or transient with loss of PRHR	LPM-REC01	core-exit T/C > 1200°F (ERG AFR.C-1)	30 minutes
	Failure to complete ADS as recovery from failure of automatic actuation or manual actuation after core damage	ADN-REC01	core-exit T/C > 1200°F (ERG AFR.C-1)	30 minutes
IS	Failure to recognize need and failure to isolate the containment, given core damage following an accident	CIC-MAN01	high containment pressure, high containment temperature, high containment radiation (ERG E-0)	~~30~~ 50 minutes
IR	Failure to recognize need and failure to open recirculation valves to flood reactor cavity after core damage	REN-MAN03	core-exit temperature > 1200°F (ERG AFR.C-1)	5 minutes
PC	Failure to recognize need and failure to open PCS water valves to drain cooling water on containment shell	PCN-MAN01	high containment pressure (ERG E-0)	**18 hours** ~~60 minutes~~
VNT	Failure to recognize need and failure to open containment vent to reduce containment pressure	VNT-MAN01	high containment pressure (SAMG)	60 minutes
IG	Failure to recognize need and failure to actuate hydrogen control system, given core damage following an accident	VLN-MAN01	core-exit T/C > 1200°F (ERG AFR.C-1)	~~10~~ 15 minutes

ATTACHMENT 43C

EVALUATION OF OPERATOR ACTIONS

The operator actions pertinent to the CET event tree nodes are listed in Table 43C-1. A comparison with the operator actions modeled in the AP600 PRA indicates that all but one operator action still have the same performance shaping factors and time windows except one, REN-~~MAN01~~MAN03.

The REN-~~MAN01~~-MAN03 time window is estimated to be shorter for the AP1000 design since higher water levels are needed in the reactor cavity, thus a longer flooding time. To compensate for the shorter time window, the action to open valves has been moved to the first step of Emergency Response Guideline (ERG) AFR.C-1. With this revision it is estimated that the A600 HEP of 3.4E-03 for this operator action is maintained for AP1000. However, two sensitivity analyses are made to study the effect of this operator action HEP being higher, namely 3.4E-02 or 0.1. The operator action affects the IWF fault tree cutsets, and thus the probabilities q2 and q20 calculated for use in the CET. The calculations are stored in sec-44iwf folder.

The results are summarized in the following table:

REN-~~MAN01~~ MAN03 HEP =	3.4E-03	3.4E-02	0.1
Q2	2.671E-09	5.088E-09	1.029E-08
Q20	2.059E-09	3.851E-09	7.712E-09
LRF	1.95E-08	2.62E-08	3.5E-08
Ceff	91.9%	89.1	85.5%

If the REN-~~MAN01~~-MAN03 failure probability were two orders of magnitude higher than the base case, the plant LRF would have been doubled, which shows that the results are somewhat sensitive to this operator action.

Table 43C-1

EVALUATION OF CET-RELATED OPERATOR ACTIONS

SUMMARY OF OPERATOR ACTIONS FOR CONTAINMENT EVENT TREE NODES

Top Event	Description of Operator Error	Event ID	Cue(s)	Time Window	AP600 Tw/Ta/Stress	AP600 HEP/Cond HEP	AP1000 HEP	Comments
DP	Failure to recognize need for post-core-uncovery RCS depress during small LOCA or transient with loss of PRHR	LPM-REC01	core-exit T/C > 1200°F (ERG AFR.C-1)	~~20~~ 30 minutes	20/15/H	1.34E-03/ 5.0E-02	1.34E-03/ 5.0E-02	
	Failure to complete ADS as recovery from failure of automatic actuation or manual actuation after core damage	ADN-REC01	core-exit T/C > 1200°F (ERG AFR.C-1)	~~20~~ 30 minutes	5/3/H	3.02E-03/ 5.0E-02	3.02E-03/ 5.0E-02	
IS	Failure to recognize need and failure to isolate the containment, given core damage following an accident	CIC-MAN01	high containment pressure, or temperature, or radiation (ERG E-0)	~~50~~ ~~30~~ 50 minutes	60/30/H	5.71E-03/ N/A	5.71E-03/ N/A	
IR	Failure to recognize need and failure to open recirculation valves to flood reactor cavity after core damage	REN-MAN03	core-exit temperature > 1200°F (ERG AFR.C-1)	5 minutes	20/10/H	3.4E-03/0.15	3.4E-03/0.15	See sensitivity analyses
PC	Failure to recognize need and failure to open PCS water valves to drain cooling water on containment shell	PCN-MAN01	high containment pressure (ERG E-0)	~~24 hours~~ ~~60 minutes~~ 18 hours	300/120/H	1.48E-04/ N/A	1.48E-04/ N/A	
VNT	Failure to recognize need and failure to open containment vent to reduce containment pressure	VNT-MAN01	high containment pressure (SAMG)	60 minutes	N/A		1.0	Not credited
IG	Failure to recognize need and failure to actuate hydrogen control system, given core damage following an accident	VLN-MAN01	core-exit T/C > 1200°F (ERG AFR.C-1)	15~~0~~ minutes	15/10/H	1.28E-03/0.5	1.28E-03/0.5	
DP	Failure to perform ADS as recovery from failure of automatic actuation or manual actuation in later phases of SGTR event	PDS6-MANADS	Late Recovery	Hours available	0.1		0.1	Screening valve

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.048 (Response Revision 1)

Question:

Please provide a quantitative assessment of the uncertainties in the reliability of in-vessel retention for the AP1000 design using the analytical approach and tools developed through the Idaho National Engineering and Environmental Laboratory (INEEL) assessment of in-vessel retention for the AP600 (i.e., J. L. Rempe, et al., "Potential for AP600 In-Vessel Retention Through Ex-Vessel Flooding," INEEL/EXT-97-00779, December 1997). This should include an assessment of the uncertainties in heat transfer, decay heat, and material property assumptions described in Appendix B of the report, and the implications of forming the alternate debris bed configurations described in Section 2.1.2 of the report. Please provide AP1000-specific probability density function results for the final bounding state (comparable to Figures 3-5 through 11 in the report) and for each alternate debris configuration. Justify that the margins to failure are sufficient to support the lower head failure assumptions used in the AP1000 PRA.

Westinghouse Response:

The calculations performed in INEEL/EXT-97-00779 show that the results of the IVR quantification were insensitive to the additional uncertainties in heat transfer correlation and material property assumptions considered by the authors. The results were sensitive to assumptions related to heat partitioning and the alternate debris bed configurations, but Westinghouse considered these assumptions to not be physically reasonable based on the core melt progression in the AP600.

Additional analyses of AP1000 debris relocation will be added to AP1000 PRA Chapter 39 as described in the response to RAI 720.088.

Design Control Document (DCD) Revision:

See the response to RAI 720.088.

PRA Revision:

See the response to RAI 720.088.

NRC Additional Comments:

The RAI requested that W provide AP1000-specific assessments for each of the alternate debris configurations identified in INEEL's review of external reactor vessel cooling for AP600. W did not provide these assessments in their response, but appears to have performed such analyses based on information they presented during a 1/24/2003 meeting with ACRS.

Revise RAI response to include alternate debris calculation described to ACRS

Westinghouse Additional Response:

INEEL Configuration A

"Configuration A" consists of a debris pool stratified in metal-over-oxide configuration. The top metal layer is thin, and the oxide debris pool is superheated and has fully developed natural circulation. The configuration with a thin metal layer produces the focusing effect, which creates a very large heat flux at the interface between the metal layer and reactor vessel wall.

Westinghouse does not consider this configuration to be applicable to the AP1000 since our analysis demonstrates that the lower plenum debris pool will contact the lower support plate and create a thick metal layer. In the transient stages before the debris contacts the lower support plate, it is either water cooled or quenched, and not a fully developed naturally circulating pool as assessed in the INEEL "Configuration A" model.

INEEL Configuration B

"Configuration B" consists of a debris pool stratified such that a thin metal layer exists between two oxide layers. Westinghouse does not consider this configuration to be physically reasonable given the density differences between the molten metal and the oxide.

INEEL Configuration C

Configuration C consists of a debris pool stratified with a dense metal layer of uranium and zirconium at the bottom of the lower plenum debris pool. The INEEL calculation appears to be in error as it apparently has put all of the decay heat into the bottom metal layer and uniformly transferred it to the vessel wall, thus concluding that the vessel will fail.

Westinghouse's assessment of the challenges to the vessel from a bottom metal layer are presented here.

The Potential for AP1000 Reactor Vessel Failure Induced by a Stratified Debris Bed with a Bottom Metal Layer during IVR

Abstract – *In-vessel retention of molten core debris (IVR) is an important severe accident management strategy employed in the AP1000 Passive Light Water Reactor. The reactor vessel is externally cooled with water, preventing vessel failure and retaining molten debris in the lower plenum. Analysis of the heat loads from a metal-over-oxide lower plenum debris bed configuration demonstrates significant margin-to-failure. However, an alternate debris bed configuration with a dense bottom metal layer below the oxide layer can be postulated to form due to debris material interactions in the lower plenum. While the mechanisms that allowed such interactions to occur are not predicted to occur during the in-vessel debris relocation predicted for the AP1000, the alternate debris bed configuration is analyzed as a low probability uncertainty in the AP1000. Two potential vessel failure modes are addressed in the analysis: failure due to high heat loads from the bottom metal layer and failure due to high heat loads caused by thinning of the top metal layer. For bounding analyses of both of these potential failure modes, the AP1000 vessel is predicted to remain intact.*

I. INTRODUCTION AND PURPOSE

In-vessel retention of molten core debris (IVR) is an important severe accident management strategy employed in the AP1000 Passive Light Water Reactor. The reactor vessel is externally cooled with water, preventing vessel failure and retaining molten debris in the lower plenum. Analysis of the heat loads from a metal-over-oxide lower plenum debris bed configuration in the AP1000 demonstrates large margin-to-failure.[1]

The Idaho National Engineering and Environmental Laboratories (INEEL) have proposed alternate lower plenum debris bed configurations that may challenge the lower head integrity.[2] One of these alternate configurations, a stratified bed with a dense bottom metal layer (Figure 1), may be postulated to form if a significant fraction of molten zirconium can react with uranium dioxide to form uranium metal.

The purpose of this paper is to investigate the viability of IVR given uncertainty in the lower debris bed configuration, in particular with respect to the formation of a dense bottom metal layer below the oxide layer.

II. MATERIALS INTERACTIONS IN THE LOWER PLENUM DEBRIS BED

Mechanisms that allow large quantities of molten zirconium to mix with molten UO_2 to produce material interactions are not predicted to occur in the AP1000 melt progression.[3] However, it can be conservatively postulated that molten oxide debris in the lower plenum chemically interacts with a fraction of the unreacted molten zirconium. The stainless steel energy-absorbing structure below the lower support plate, which will be subsumed by oxide debris in the lower plenum, may also be postulated to chemically interact with the oxide to form uranium.

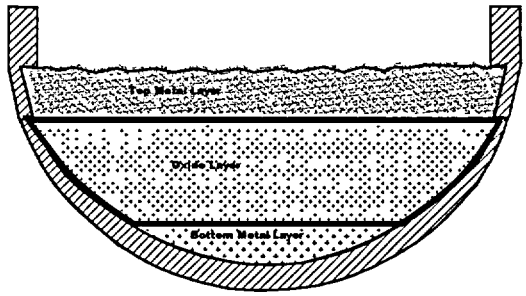


Figure 1 – Bottom Metal Layer Lower Plenum Debris Bed Configuration

The material interactions between the molten uranium dioxide and molten zirconium and steel may create uranium metal that mixes with remaining zirconium and steel to produce a dense metal debris layer that forms below the oxide layer in the lower plenum debris bed (Figure 1).

The presence of a bottom metal layer may potentially increase the challenge to the reactor vessel integrity. The metal is more conductive than the oxide. If a significant fraction of the decay heat bearing fission products is partitioned into the metal, the heat flux to the vessel wall at the very bottom of the lower head may exceed the critical heat flux at the external surface of the vessel. Additionally, the loss of zirconium mass from the top metal layer to the bottom layer will thin the top metal layer and therefore increase the heat flux from the top metal layer to the vessel wall via the focusing effect.[4]

This paper presents an analysis quantifying, in a bounding manner, challenges to the lower head integrity from a bottom metal layer and a thinned top metal layer produced by material interactions in the debris bed.

II. BOTTOM METAL LAYER HEAT TRANSFER MODEL

An analysis to quantify the heat flux from a bottom metal layer in the AP600 reactor vessel was performed by the Idaho National Engineering and Environmental Laboratory (INEEL).[2] The bottom metal layer in the debris bed configuration is 0.5-meter high mixture of uranium (40 weight percent) and zirconium, which is volumetrically heated by fission products and actinides. Based on the INEEL calculation, the heat flux from the bottom metal layer to the vessel wall is a constant heat flux of approximately 4300 kW/m^2, and the report concludes that reactor vessel failure is predicted for this debris bed configuration.

However, note that a 0.5-meter deep pool in a 2-meter hemispherical plenum has a surface area on the inside vessel lower head of 6 m^2. Therefore, total rate of heat transferred to the vessel wall from the bottom metal layer in the INEEL calculation is 26 MW. At 1.5 hours, the earliest time that such a configuration is expected, the total decay heat in the AP600 debris, with the heat from volatile fission products removed, is less than 20 MW. Therefore, there is too much heat modeled in the bottom metal layer debris in the INEEL analysis.

The temperature difference required to conduct a uniform heat flux of 4300 kW/m^2 through a volumetrically heated, 0.5 meter thick metal layer is on the order of tens of thousands of degrees Kelvin. Therefore, the INEEL model is not a physically reasonable representation of the heat transfer phenomena associated with the bottom metal layer, and over-predicts the heat flux from the bottom metal layer to the vessel wall.

A thick bottom metal layer cannot conduct a significant heat load through the layer to the vessel wall. If a large temperature difference develops in the metal layer to conduct heat to the wall, the metal temperature will exceed the oxide temperature and a fraction of the decay heat will be transferred upward into the oxide layer. Thus, a thick bottom metal layer that is volumetrically heated by a significant fraction of the decay heat will develop a temperature profile with a peak temperature in the middle of the pool. The layer will stratify into two sublayers (see Figure 2) and heat will be transferred both upward and downward. The bottom sublayer will transfer heat downward via conduction through the layer to the vessel wall and the top sublayer will transfer heat upward via convection to the upper surface, the interface with the oxide layer.

The actual heat transfer in the lenticular geometry of the bottom metal layer is complex. To model the system in detail would require computational fluid dynamics (CFD) modeling or additional testing programs. For the analysis presented here, a conservative approach with a simplified model is taken.

To find the maximum heat flux from the bottom metal layer to the vessel wall at the point with the least margin-to-failure, the very bottom of the vessel, the problem can be conservatively approximated as an infinite slab of molten metal of thickness H. The infinite slab model is comprised of two sublayers with an adiabatic boundary between them (Figure 2). The bottom conduction sublayer has a thickness, H_{bot}, and the top convection sublayer has a thickness of H_{top}.

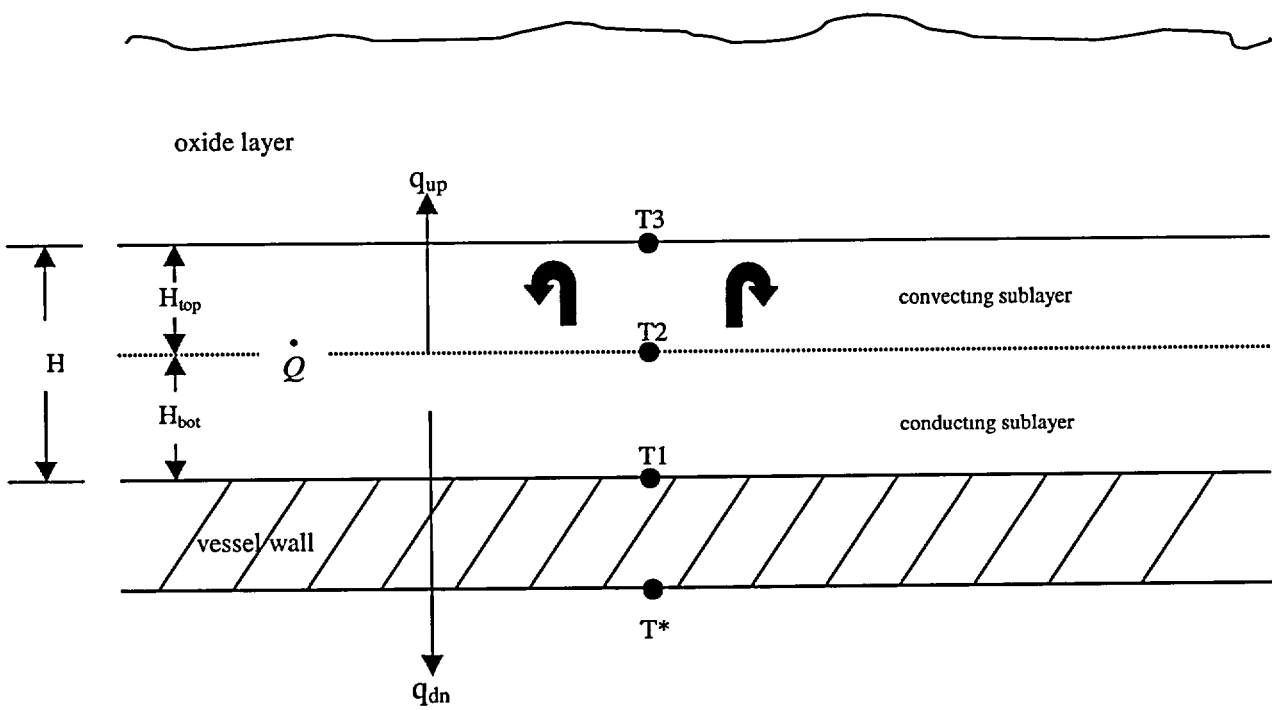


Figure 2 – Model for Lower Plenum Bottom Metal Layer Heat Transfer

The bottom conducting sublayer is volumetrically heated and conducts heat downward through the sublayer to the vessel wall such that:

$$T_2 - T_1 = \frac{\dot{Q}_m H_{bot}^{\ 2}}{2K_m} \qquad (1)$$

The conduction through the vessel wall to the water-cooled outer surface is expressed as:

$$q_{m-dn} = \dot{Q}_m H_{bot} = K_W \frac{T_1 - T^*}{x_W} \qquad (2)$$

In equations 1 and 2, the inner wall temperature T_1 has a maximum corresponding to the liquidus temperature in the U-Zr-Fe eutectic. If the wall is melting at the heat flux $q_{m\text{-}dn}$, the thickness of the wall is found from equation 2, given the maximum value of T_1.

The heat transfer in the upper convection layer is described by the Nusselt number (Nu) correlation developed by Theofanous-Angelini (T-A correlation) for a pool with an adiabatic lower boundary cooled from the top.[5]

$$Nu_{up} = 0.206 Ra^{0\,303} \Pr^{0\,084} \qquad (3)$$

The T-A correlation can be translated[5] from the external Rayleigh number (Ra) to the internal Rayleigh number (Ra') by noting that for the geometry, the Damkőhler number (Da) is exactly equal to the Nusselt number and that Ra' ≡ Ra * Da.

$$Nu_{up} = 0.297 Ra'^{0\,233} \Pr^{0\,0645} \qquad (4)$$

The height (H_{top}) of the convection layer is related to equation 4 through Ra'.

$$Ra' = \frac{g\beta \dot{Q}_m H_{top}^{\ 5}}{K_m \nu \alpha} \qquad (5)$$

The heat transfer coefficient to the oxide pool, h_{m_up}, is found from Nu_{up}.

$$h_{m\_up} = Nu_{up} * \frac{K_m}{H_{top}} \qquad (6)$$

And the upward heat flux, $q_{m\text{-}up}$ is

$$q_{m-up} = h_{m-up}(T_2 - T_3) \qquad (7)$$

Table 1 - Properties of the Bottom Metal Pool used in the Analyses				
Property	Zr	SS	U	Pool
mass (kg)	5566	2277	5224	13066
volume (m^3)	0.908	0.324	0.299	1.531
density (kg/m^3)	6130	7020	17500	8536
thermal conductivity (W/m-K)	36.0	24.1	49.0	34.3
specific heat (J/kg-K)	458	835	157	530
volumetric expansion (K^{-1})	5.4E-05	1.2E-04	8.6E-04	2.2E-04
viscosity (Pa-s)				1.1E-03
thermal diffusivity (m^2/s)				7.59E-06
kinematic viscosity (m^2/s)				1.29E-07
Prandtl number				1.70E-02
molecular weight (g/mole)	91.2	55.1	238.0	
mole fraction	4.91E-01	3.32E-01	1.77E-01	

The properties of the metal pool are evaluated at the bulk temperature of the convection sublayer.[5]

The calculation is performed by iterating on the height of H_{bot} and noting that the sum of H_{bot} and H_{top} is equal to the total maximum thickness of the bottom metal layer, converging on the common temperature at the boundary between the two layers, T_2, in equations 1 and 7.

III. SUCCESS CRITERION

The reactor vessel will remain intact if the surrounding water can cool the external surface of the vessel wall via nucleate boiling.[4] The critical heat flux at the azimuthal angular position, (0° degrees at the bottom of the vessel and 90° at the top of the hemisphere) defines the limiting heat flux that can be transferred from an intact vessel wall. Therefore, if the heat flux from the debris to the reactor vessel wall is predicted to be less than the critical heat flux, the reactor vessel integrity is maintained. The critical heat flux on the AP1000 reactor vessel external surface is defined by the results of the ULPU Configuration-IV tests.[1]

IV. COMPOSITION OF THE BOTTOM METAL POOL

The composition of the bottom metal pool is calculated based on the following bounding assumptions that maximize the thickness of the layer:

- The entire mass of the stainless steel energy absorbing structure (3000 kg) in the lower plenum reacts with relocated uranium dioxide to produce uranium, iron and iron oxide. The other components of the stainless steel are neglected.

- 7000 kg of zirconium (31 percent of the total zirconium in the core) that is molten during the oxide relocation to the lower plenum mixes with the oxide and reacts with the uranium dioxide to form uranium, zirconium, and zirconium dioxide.

- The fraction of uranium in the metal layer has a maximum of 40 weight percent, based on peer review comments by Olander,[5] and consistent with the assumption in the INEL calculation.[4]

Employing these assumptions, the bottom metal pool has the properties outlined in Table 1. The total height of the 1.53 m^3 pool in the lower plenum is 0.58 m.

IV. VOLUMETRIC HEAT RATE OF THE BOTTOM METAL LAYER

Heat in the metal layer is produced by the decay of fission products and actinides partitioned into the metal layer during its formation. Only a fraction of metallic fission products are expected to be in the metal layer. For this calculation, it is conservatively assumed that the decay heat partitioned into the metal layer is one hundred percent of the total decay heat associated with the equivalent volume of oxide material reacted to produce the uranium.

The volumetric heat density of the bottom metal layer is found from the equation:

$$\dot{Q}_m = \frac{m_u * \left(\frac{mw_{UO2}}{mw_U}\right)}{\rho_{ox} * V_m} * \dot{Q}_{ox} \qquad (8)$$

Assuming an upper bound volumetric heat density of the oxide initially relocated to the lower plenum of 3.0 MW/m^3, the heat density in the bottom metal layer is 1.38 MW/m^3, or approximately 6 percent of the total decay heat in the debris bed.

V. CALCULATION OF THE BOTTOM METAL POOL HEAT FLUXES

If the bottom metal layer is transferring heat to the oxide layer, the interface temperature between the bottom metal pool and the oxide layer must be greater than the melting temperature of the oxide. Therefore, there is no oxide crust between the layers and heat transfer behavior of the two pools is coupled. For this calculation, the interface temperature between the pools is assumed to be 3250°K, a high estimate of the superheat temperature of the oxide pool. The high estimate is conservative since it forces more heat to go out through the vessel wall, thus increasing the heat flux to the wall.

The inside surface of the vessel wall is melting, and therefore, the temperature of the wall is the liquidus temperature of the U-Zr-Fe eutectic, which is initially assumed to 1600°K and investigated further with a sensitivity case.

Solving the equations to converge on T2 produces the following results: The lower conduction sublayer of the bottom metal pool is 30 cm thick. The heat flux from the bottom metal layer to the vessel wall is 415 kW/m^2. The inside of the vessel wall is melted and the remaining thickness of the wall is 12 cm. The upper convection layer is 28 cm thick and has a heat flux of 389 kW/m^2 to the oxide layer.

If the U-Zr-Fe eutectic temperature is assumed to be 1400°K, the melting temperature of uranium and a reasonable lower bound for the eutectic liquidus temperature, the thickness of the lower conduction sublayer is 32 cm and the heat flux is 437 kW/m^2. The remaining vessel wall thickness is 11 cm. Thus, the results are not particularly sensitive to this melting temperature.

VI. THINNING OF THE TOP METAL LAYER

The interaction of metal and oxide produces a second potential failure mechanism for IVR by increasing the heat flux from the metal layer to the reactor vessel wall. The loss of zirconium and steel to the bottom metal layer thins the top metal layer, and the focusing effect[4] will increase the heat flux to the vessel wall from the top metal layer. The heat load from the thinned top metal layer is conservatively quantified in this section.

The assumptions for this calculation maximize the heat flux from the top metal layer to the vessel wall by minimizing the metal mass available to form the layer and maximizing the heat transfer from the oxide layer to the top metal layer:

- 3000 kg of stainless steel participates in the interaction and is removed from mass of metal available to form the top metal layer. The mass of steel available to form the top metal layer is limited to 47000 kg, a lower bound value.
- The relatively small mass of iron oxide is neglected in the calculation of the oxide layer properties.
- 7000 kg of zirconium (31 percent of the total zirconium mass) participates in the interaction and contributes 5566 kg of zirconium to the bottom metal layer. The remaining mass of zirconium in the core is assumed to be totally oxidized by steam and by the interaction with UO_2. No zirconium is included in the formation of the top metal layer.
- The material interaction produces 5224 kg of uranium, which removes 5926 kg of UO_2 from the oxide layer.
- All of the decay heat is in the oxide layer. None of the fission product heating is lost to the bottom metal layer.

The heat flux from the bottom of the vessel to the top of the debris pool is quantified using the methodology developed in the AP600 IVR Analysis.[5] The critical heat flux from ULPU Configuration IV tests[1] is the success criterion. The results of the calculation are presented in Figures 3 and 4.

The heat flux from the metal layer is 1720 kW/m^2. The minimum margin to failure occurs at the bottom of the metal layer, 84° azimuthally, where the critical heat flux is 1890 kW/m^3. The q/qchf is 0.91. Therefore, even for this bounding case of metal layer thinning, there is still margin to vessel failure.

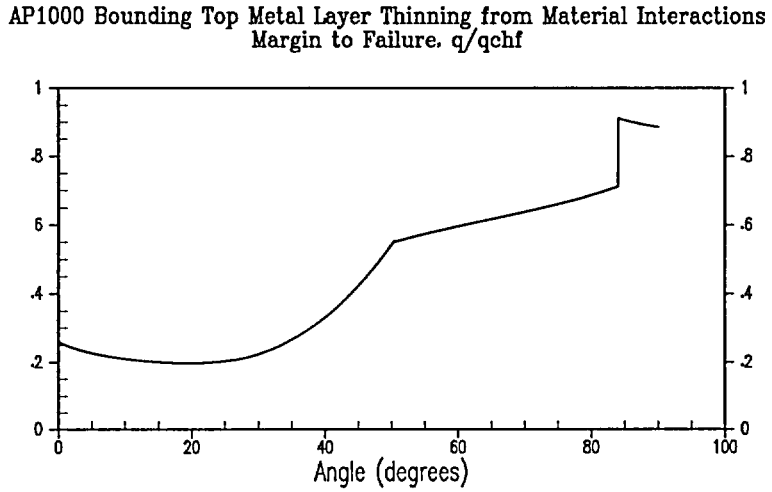


Figure 3 – Heat Fluxes from Debris Bed with a Thinned Metal Layer to the Reactor Vessel Wall

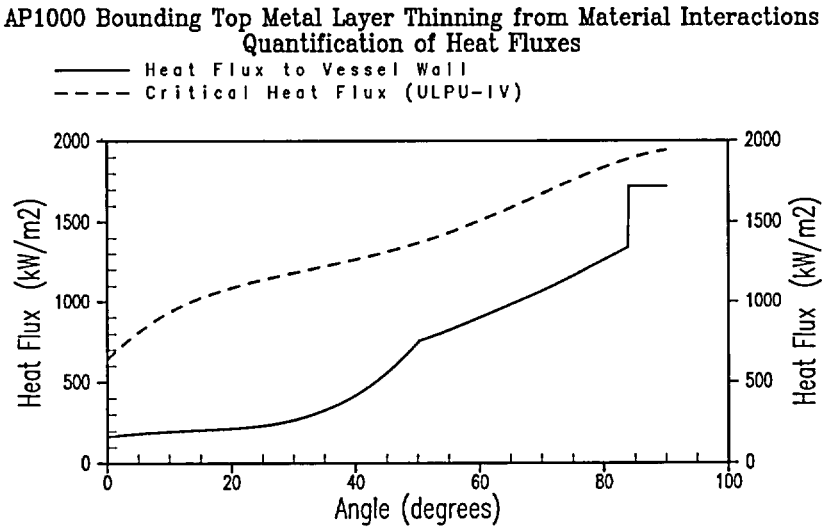


Figure 4 – Margin-to-Failure from Debris Bed with a Thinned Metal Layer to the Reactor Vessel Wall

VII. CONCLUSIONS

The analysis presents bounding cases for challenges to the vessel integrity from two potential failure modes postulated from phenomena associated with lower plenum material interactions.

The first potential failure mode is a challenge from high heat fluxes generated by a bottom metal layer with a significant fraction of the fission products partitioned into the metal. The lower bound critical heat flux[1] at the bottom point of the reactor vessel lower head for the AP1000 geometry is greater than 800 kW/m^2. The upper bound of the heat flux from the bottom metal layer to the vessel wall is predicted to be less than 500 kW/m^2. Therefore, the heat flux from a bottom metal pool is not expected to exceed the critical heat flux at the bottom of the reactor vessel lower head. The ratio of the maximum heat flux to the minimum critical heat flux, q/qchf is 0.625.

The second potential failure mode is from a thinned metal layer producing high heat fluxes via the focusing effect. For a bounding case of top metal layer thinning due to material interactions, the peak heat flux in the metal layer is 1720 kW/m^3. The lower bound critical heat flux in the metal layer at the azimuthal angle is 1890 kW/m^3. The q/qchf is 0.91, demonstrating that margin-to-failure is maintained for this bounding case.

It is noted that actual heat fluxes expected from the debris bed to the lower vessel wall, even in the presence of material interactions and a bottom metal layer are significantly less than predicted in this analysis. The assumptions in this analysis are conservative with respect to increasing the heat loading to the vessel. Mechanistic assumptions are expected to result in milder thermal loading and larger margins-to-failure.

Therefore, the AP1000 reactor vessel is predicted to remain intact during IVR considering bounding conditions generated from material interactions postulated to occur in the lower plenum debris bed.

NOMENCLATURE

α = thermal diffusivity of the convection sublayer (m^2/sec)
β = volume thermal expansion coefficient in the convection sublayer (1/K)
Da = Damköhler number of the convecting sublayer of the bottom metal layer.
g = acceleration of gravity (9.81 m/sec^2)
H_{top} = thickness of convection layer (m)
H_{bot} = thickness of conduction layer (m)
$h_{m\text{-}up}$ = heat transfer coefficient in the convection sublayer (W/m^2-K)
K_m = thermal conductivity of bottom metal layer (W/m-K)
K_W = reactor vessel wall thermal conductivity (W/m-K)
m_u = mass of uranium in the bottom metal layer (kg)
mw_{uo2} = molecular weight of UO_2 (g/mole)
mw_U = molecular weight of uranium (g/mole)
ν = kinematic viscosity of the convection sublayer (m^2/s)
Nu_{up} = Nusselt number of the top convection sublayer
ρ_{ox} = density of the oxide debis (kg/m^3)
Pr = Prandtl number of the convection sublayer

$\dot{Q}_{ox}$ = volumetric heat density of oxide debris (W/m^3)

$\dot{Q}_m$ = volumetric heat density in bottom metal layer (W/m^3)

$q_{m\text{-}dn}$ = downward heat flux from the bottom metal layer to vessel wall (W/m^2)

$q_{m\text{-}up}$ = upward heat flux from the bottom metal layer to the oxide layer (W/m^2)

Ra = external Rayleigh number of the convecting sublayer of the bottom metal layer

Ra' = internal Rayleigh number of the convecting sublayer of the bottom metal layer

T_1 = temperature at the bottom metal layer / vessel wall interface (K)

T_2 = temperature at the boundary between conduction and convection layers (K)

T_3 = temperature at the bottom metal layer / oxide layer interface (K)

T^* = the vessel outer surface temperature ~ T_{sat}

V_m = volume of the bottom metal layer (m^3)

x_W = thickness of the vessel lower head wall (m)

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2. J. L. Rempe, D. L. Knudson, M. Cebull, C. L. Atwood, Potential for In-Vessel Retention through Ex-Vessel Flooding, pp.365-374, Proc. OECD/CSNI Workshop on In-Vessel Core Debris Retention and Coolability, Garching, Germany, 3rd-5th March, 1998.

3. AP1000 PRA Report, Revision 1, Westinghouse Electric Company, March 2003.

4. Theofanous, T. G., et. al., In-Vessel Retention and Coolability of a Core Melt, DOE/ID 10460, July1995.

5. Theofanous, T. G., Angelini, S., Natural Convection for In-Vessel Retention at Prototypic Rayleigh Numbers, Nuclear Engineering and Design 200 (2000) 1-9.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.050 (Response Revision 1)

Question:

Westinghouse claims in Chapter 5.3.5.4 of the AP1000 DCD that the forces on the AP1000 reactor vessel insulation following core relocation and cavity flooding can be based on AP600 test results from the ULPU-2000 test program for Configuration III. Although this test data was used to develop the functional requirements for the AP600 reactor vessel insulation and support system, its suitability and applicability for AP1000 has not been established, and is questionable given the substantial differences between the AP600 and AP1000 insulation system designs and accident conditions. The AP1000 design would have higher heat fluxes from the vessel, higher water/steam flow rates and flow velocities through the insulation system, a considerably smaller gap between the insulation and reactor vessel, and closely-fitted hemispherically-shaped insulation panel (versus conically-shaped insulation with a substantial standoff distance from the reactor vessel in AP600). Collectively, these differences could result in substantially different pressure loads and functional requirements for the AP1000 reactor vessel insulation and support system. Westinghouse needs to either: (1) establish the applicability of the ULPU Configuration III test results to AP1000 considering the impact of each of the above factors, or (2) develop AP1000-specific test data based on the prototypical insulation and flow conditions for AP1000, i.e., ULPU-2000 Configuration IV. Note: Westinghouse also states in Chapter 5.3.5.4 that further evaluation of the forces on the reactor vessel insulation and supports is provided in the AP1000 PRA. Such information is not provided in the PRA, e.g., in Chapter 39 "In-Vessel Retention of Molten Core Debris."

NRC Additional Question:

The RAI requested that Westinghouse either: (1) establish the applicability of the ULPU Configuration III test results to AP1000, or (2) develop AP1000-specific test data based on the prototypical insulation and flow conditions for AP1000. This was not addressed in the response.

Revise response to clarify COL applicant requirement to use ULPU Configuration V.

Westinghouse Response Revision 1:

The structural analysis of the AP1000 reactor vessel insulation support frame has not been completed. The analysis will be completed by the COL applicant. A new COL item has been added to the DCD **and is included in DCD Revision 3.** The structural loads are expected to be similar to the AP600 loads and no feasibility issues exist for the structural design.

As stated, the insulation design for AP1000 is different than AP600, and the loads on the insulation are expected to be different than those derived from the ULPU-2000 Configuration III tests. It is expected that the hydrostatic loads on the insulation will be similar to those measured for AP600. The dynamic loads from boiling are expected to be higher due to the

higher heat flux in the coolant channel. These calculations will be compared to test data to assure that the insulation and supports are adequately designed for severe accident loads.

The DCD will be revised to clarify that the ULPU Configuration IV and Configuration V test data is suitable to develop the design loads for the AP1000 reactor vessel insulation design.

Design Control Document (DCD) Revision:

DCD Revision 3 (which incorporated the original RAI response) will be revised as follows to address the NRC additional comment:

5.3.6.5 Reactor Vessel Insulation

The Combined License applicant will address verification that the reactor vessel insulation is consistent with the design bases established for in-vessel retention. **The ULPU Configuration IV and Configuration V test data is suitable to be used to develop the design loads for the AP1000 reactor vessel insulation design.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.060 (Response Revision 1)

NRC Additional Comments:

The original RAI 720.060 requested that Westinghouse perform a SAMDA evaluation for the AP1000, consistent with the scope and content that was performed for the AP600 SAMDA evaluation. The NRC has reviewed the AP1000 SAMDA evaluation provided in the Westinghouse response to RAI 720.060 and has the following additional comments provided below. These comments were discussed at the meeting held between members of the NRC and Westinghouse at NRC headquarters on February 27, 2003.

The evaluation of SAMDAs was omitted from the PRA/DCD and submitted in response to this RAI. The evaluation does not address a number of items called out in the RAI and has several additional deficiencies, as summarized below:

1. The cost benefit methodology appears to be based on an outdated guidance document (NUREG/CR-3568, 1983). The current guidance for regulatory analysis contained in NUREG/BR-0184 (1997) and NUREG/BR-0058 (2000) should be applied.
2. Replacement power costs were omitted. These averted onsite costs need to be included consistent with SECY-99-169.
3. The CDF and population dose values used in the evaluation only reflect internal events. The contribution to CDF and population dose from shutdown and fire events should also be included.
4. The RAI requested an explanation of how insights from the AP1000-specific PRA and supporting risk analyses for external and shutdown events, including importance analyses and cutset screening, were used to identify potential plant improvements. This was not addressed in the response.
5. The RAI requested justification that the potential improvements identified through a systematic process (as suggested above) are included within the set of 15 SAMDAs identified in Appendix 1B of the AP1000 DCD. This was not addressed in the response.

Westinghouse Revised Response:

Response to Questions 1, 2, and 3:

The response to questions 1, 2, and 3 is incorporated into section 7 of Attachment 1; namely, the method suggested in NUREG/BR-0184 is used; replacement power costs are included; the

contribution of shutdown and fire events is included. The conclusions of the previous submittal did not change.

Response to Question 4:

Insights from the AP1000-specific PRA and supporting risk analyses for external and shutdown events, including importance analyses and cutset screening were used continuously throughout the development of the AP600/AP1000 design in the decade extending from 1990 through the present. The PRA dominant cutsets and sequences were monitored to identify the risk contributors and if feasible alternatives (design, procedural, tech-spec, etc.) to reduce the plant risk were identified, they were implemented. The total plant risk has been lowered to a point (See the response to questions 1-3 above), that it has become very difficult to find further cost-effective plant improvements. This design improvement process has been an intimate and natural part of the design and risk evaluation process.

The following examples of more recent use of PRA insights for AP1000 design are listed to illustrate the type of activities that have actually occurred:

1. Passive containment cooling system reliability is increased in response to the uncertainty in the success criteria of air-alone containment cooling, by including a third diverse valve.
2. Different AC power sources are assigned to some RNS components to eliminate the support system failures that could fail startup feedwater and main feedwater systems from also failing the RNS system (this is relevant in transient sequences, not so important for LOCA events).
3. The explosive valve types for the sump recirculation lines have been changed to reduce the common cause failures in the recirculation system.
4. The probability of success of operator action, after core damage occurs, to perform reactor cavity flooding is increased by moving the action to the beginning of the functional restoration guidelines.

Response to Question 5:

The SAMDA alternatives were chosen for AP600 by examination of the SAMDA alternatives generally considered in the industry, plus expert-opinion based alternatives that may be specific to the AP600/AP1000 designs. These SAMDA alternatives are also considered for AP1000 design, due to its similarity to the AP600 design.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Attachment 1

AP1000 SAMDA EVALUATION

1 Introduction

This response provides an evaluation of Severe Accident Mitigation Design Alternatives (SAMDA) for the Westinghouse AP1000 design. This evaluation is performed to evaluate whether or not the safety benefit of the SAMDA outweighs the costs of incorporating the SAMDA in the plant, and is conducted in accordance with applicable regulatory requirements as identified below.

The National Environmental Policy Act (NEPA), Section 102.(C)(iii) requires, in part, that

...all agencies of the Federal Government shall ... (C) include in every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on ... (iii) alternatives to the proposed action.

10 CFR 52.47(a)(ii) requires an applicant for design certification to demonstrate

> ... compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f) ...

A relevant requirement of 10 CFR 50.34(f) contained in subparagraph (1)(i) requires the performance of

> ... a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant ...

In SECY-91-229, the NRC staff recommends that severe accident mitigation design alternatives be addressed for certified designs in a single rulemaking process that would address both the 10 CFR 50.34 (f) and NEPA considerations in the 10 CFR Part 52 design certification rulemaking. SECY-91-229 further recommends that applicants for design certification assess SAMDAs and the applicable decision rationale as to why they will or will not benefit the safety of their designs. The Commission approved the staff recommendations in a memorandum dated October 25, 1991 (Reference 1).

2 Summary

Note that AP1000 is very similar to AP600, which has received Design Certification. The evaluation for AP1000 uses the conclusions of the AP600 SAMDA investigation as described below. An evaluation of candidate modifications to the AP600 design was conducted to evaluate the potential for such modifications to provide significant and practical improvements in the radiological risk profile of the AP600 design. Since AP1000 is so similar to AP600, the list of candidate modifications is the same.

The process used for identifying and selecting candidate design alternatives included a review of SAMDAs evaluated for other plant designs. Several SAMDA designs evaluated previously for other plants were excluded from the present evaluation because they have already been incorporated or otherwise addressed in the AP600 and AP1000 designs. These include:

Hydrogen ignition system
Reactor cavity flooding system
Reactor coolant pump seal cooling
Reactor coolant system depressurization
Reactor vessel exterior cooling.

Additional design alternatives were identified based upon the results of the AP600 probabilistic risk assessment (Reference 3). The AP1000 probabilistic risk results are similar to those developed for AP600. Fifteen candidate design alternatives were selected for further evaluation.

An evaluation of these alternatives was performed using a bounding methodology such that the potential benefit of each alternative is conservatively maximized. As part of this process, it was assumed that each SAMDA performs beyond expectations and completely eliminates the severe accident sequences that the design alternative addresses. In addition, the capital cost estimates for each alternative were intentionally biased on the low side to maximize the risk reduction benefit. This approach maximizes the potential benefits associated with each alternative.

The results show, for AP600 and AP1000, that despite the significant conservatism employed in the evaluation, none of the SAMDAs evaluated provide risk reductions which are cost beneficial. The results also show that even a conceptual "ideal SAMDA", one which reduces the total plant radiological risk to zero, would not be cost effective. This is due primarily to the already low risk profile of the AP600 and AP1000 designs.

3 Selection of SAMDAS

Candidate design alternatives were selected based upon design alternatives evaluated for other plant designs (References 4, 5, and 6) as well as suggestions from AP600 and AP1000 design personnel. Additional candidate design alternatives were selected based upon an assessment of the AP600 and AP1000 probabilistic risk assessment results. Fifteen design alternatives were finally selected for further evaluation. These fifteen SAMDAs are:

- Chemical volume and control system (CVS) upgraded to mitigate small LOCAs
- Filtered containment vent
- Normal residual heat removal system (RNS) located inside containment
- Self-actuating containment isolation valves
- Passive containment spray
- Active high pressure safety injection system
- Steam generator shell side passive heat removal system
- Steam generator safety valve flow directed to in-containment refueling water storage tank (IRWST)
- Increase steam generator secondary side pressure capacity
- Secondary containment filtered ventilation
- Diverse IRWST injection valves
- Diverse containment recirculation valves
- Ex-vessel core catcher
- High pressure containment design
- Diverse actuation system (DAS) improved reliability.

A description of each design alternative evaluated for AP600 is presented in Appendix A to this document.

4 Methodology

The severe accident mitigation design alternatives analysis employs a bounding methodology such that the benefit is conservatively maximized and the capital cost is conservatively minimized for each SAMDA.

4.1 Total Population Dose

To assess the potential benefits associated with a design alternative, estimates are made of the total offsite population dose resulting from each of the release categories (i.e., source terms). MACCS2 version 1.12 (Reference 9) is used for the analysis. The NRC sponsored the development of this code. The code performs

probabilistic estimates of offsite consequences from potential accidental releases in conformance with Chapter 9 of the probabilistic risk assessment (PRA) guidelines described in NUREG/CR-2300 (Reference 10).

Doses are determined for the early exposure effects resulting from the initial 24 hours following the core damage initiation. The dose evaluation provides the conditional probability distributions for the consequence measures, which includes the whole-body dose for this analysis. These consequence probability distributions are based on the assumption that the accident that produced the source term has occurred. Therefore, the consequence probability distributions presented result from the variation in dose levels due to the various meteorological conditions. Hence, the actual probability of the identified dose levels would be the probability of the release category that produced the source term occurring multiplied by the probability of the dose level.

The dose risks are quantified by multiplying the calculated fission product release category frequency vector by the release category mean dose vectors. The frequencies for each of the six release categories are quantified in Chapter 45 of the AP1000 PRA (Reference 2), while the mean doses for each release category are identified in revised Chapter 49 (see response to RAI 720.056). Table B-1 presents the results of the dose risk calculations at the site boundary for 2 hours of exposure. The table presents the release category identifier, the release frequency (per reactor-year), the mean dose (in rem), and the resulting risk (in rem per reactor-year). In addition, each table presents the total dose risk and the percent that each release category contributes to the total risk.

It is shown that release category CFE presents the largest risk to the site safety.

4.2 AP1000 RISK (CDF, LRF, and POPULATION Dose)

Summary of AP1000 Risk (CDF and LRF)		
	CDF	LRF
Internal Events at Power	2.41E-07/yr	1.95E-08/yr
Events at Shutdown	1.23E-07/yr	2.05E-08/yr (2)
Internal Fire	5.61E-08/yr	4.54E-09/yr (2)
Internal Flooding	8.82E-10/yr	negligible
Seismic Events	not quantified (1)	not quantified (1)

Notes:
(1) = Seismic margins method is used. CDF and LRF not quantified.
(2) = LRF is not quantified, but is estimated by a ratio of CDF to LRF for corresponding cases: namely AP600 for shutdown, internal events for fire.

Level 3 analysis is performed only for internal events at power. The ensuing population dose was very low and it was not pursued for other events. The population dose for internal events is given below, in Table 4-1.

Table 4-1

POPULATION WHOLE BODY DOSE (EFFECTIVE DOSE EQUIVALENT, EDE), 0-80.5 KM PERSON-SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	7.88E+03	6.11E+03	1.47E+04	2.01E+04	3.21E+04	3.51E+04	5.34E+04
CFE	8.51E+03	6.25E+03	1.62E+04	2.31E+04	4.13E+04	5.06E+04	6.40E+04
DIRECT	2.16E+01	1.20E+01	4.78E+01	8.13E+01	1.14E+02	1.23E+02	1.68E+02
IC	7.19E+00	4.21E+00	1.71E+01	2.95E+01	3.56E+01	3.84E+01	5.60E+01
BP	2.91E+03	1.74E+03	5.90E+03	1.00E+04	1.52E+04	1.81E+04	2.58E+04
CI	2.01E+04	1.13E+04	4.71E+04	6.60E+04	1.23E+05	1.48E+05	1.61E+05
CFL	5.32E+03	3.87E+03	1.04E+04	1.35E+04	2.32E+04	2.77E+04	4.35E+04
72-hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CFI	8.89E+03	6.89E+03	1.63E+04	2.21E+04	3.42E+04	3.84E+04	5.73E+04
CFE	9.36E+03	6.89E+03	1.88E+04	2.54E+04	4.25E+04	5.12E+04	6.77E+04
DIRECT	2.45E+01	1.43E+01	5.50E+01	8 33E+01	1.16E+02	1.26E+02	1.78E+02
IC	8.80E+00	5.57E+00	1.98E+01	3.14E+01	4.41E+01	5.03E+01	6.33E+01
BP	3.11E+03	1.85E+03	6.31E+03	1.03E+04	1.54E+04	1.82E+04	2.69E+04
CI	2.14E+04	1.25E+04	4.90E+04	7.40E+04	1.27E+05	1.53E+05	1.67E+05
CFL	5.84E+03	4.32E+03	1.12E+04	1.48E+04	2.53E+04	3.04E+04	4.62E+04

5. A Summary of Risk Significant Enhancements

5.1 Use of PRA Insights and Design Improvements

Three specific discussions of how Insights from the AP1000 PRA and supporting analyses are given in existing documents as follows:

1. Response to RAI 720.040 lists and discusses the PRA insights that impacted the design;

2. AP1000 RTNSS WCAP-15985 "AP1000 Implementation of the regulatory Treatment of Nonsafety-Related System Process" uses additional insights to classify some non-safety systems to be considered for Technical Specifications and Administrative Controls (see response to RAI 720.039);

3. Reliability assurance program (RAP) (DCD Section 17.4) uses component importances and expert input to identify those components that are risk significant.

Documents for items 2 and 3 can be consulted for further details. Item 1 is summarized below.

5.2 AP1000 Design Enhancements Already Implemented

This section summarizes the design enhancements that are already incorporated into the AP1000 plant due to PRA insights and results.

1. Changed the normal position of the two Containment motor operated recirculation valves (in series with squib valves) from closed to open

The normal position of the two MOV lines in the two sump recirculation lines have been changed from NORMALLY CLOSED to NORMALLY OPEN to improve the reliability of opening these paths. These 2 paths support containment recirculation for core cooling and IRWST draining for IVR. This change reduced the CDF and LRF contribution from the failure modes to open the MOVs.

2. Changed IRWST drain procedure so it occurs earlier for IVR support

Credit is taken for operator action to drain the IRWST into the sump to preserve reactor vessel integrity following core melt. The procedure for this severe accident response has been modified so that the operator action associated with IRWST draining is moved to the beginning of the procedure to allow more time for operator success and also to fill the cavity as soon as possible. This improves the probability of success of the operator action.

3. Improved IVR heat transfer

In going from AP600 to AP1000, the heat loads during IVR are increased due to the larger core power level which reduced the margins in the heat removal capability through the reactor vessel head during IVR . To compensate for the increase in core power, the critical heat flux limit on the outside of the reactor vessel has been increased by changes made to the flow path between the outside of the reactor

vessel and the reactor vessel insulation. Testing has confirmed the robustness of the IVR heat transfer.

4. Improved IRWST vents

The larger core in the AP1000 can generate more hydrogen in a severe accident. In the AP1000 hydrogen analysis for Level II, it was observed that the standing hydrogen diffusion flames at the IRWST vents resulted in a larger thermal loads to the containment steel shell, potentially leading to containment wall failure. The design of the vents were changed so that the IRWST vents located well away from the containment would open and the IRWST vents located next to the containment would not open during a severe to eliminate or minimize this potential concern.

5. Incorporated low boron core (ATWT)

In AP600, ATWS contribution to LRF was noticed to be high relative to other initiating events. A low boron core was incorporated into the design to reduce the potential contribution of ATWS to plant risk.

6. Added 3rd Passive Containment Cooling drain valve (MOV diverse to AOV)

Due to reduced containment surface area per MW of core power, natural air circulation without PCS water drain may not always be sufficient for long term (> 1 day) containment heat removal in AP1000. For AP600 it was always sufficient for an indefinite time. To reduce the uncertainty in whether air cooling is sufficient to provide adequate long-term containment heat removal, a third path was added to the PCS drain lines to increase PCS reliability. The isolation valve used in the third path is an MOV, which is diverse from the AOVs used in the other two lines. This provides considerable improvement in the PCS water drain reliability.

7. Reduced Potential Recirculation-Line Squib Valve Failures

An examination of AP1000 plant CDF cutsets revealed that the CCF of 4/4 recirculation line squib valves is a dominant contributor to CDF and LRF. This failure mode can be reduced by re-aligning the diverse squib valves already used in the AP1000 (and AP600) IRWST injection paths (high pressure valves) and the containment recirc paths (low pressure valves). By making the recirculation squib valves two sets of two LP and HP squib valves, which are different and belong to different CCF groups. This design change reduces the CCF failure contribution of the recirculation squib valves. The increase in the group size of the HP squib valves from 4 to 6 (including the four from the IRWST injection lines) does not add an appreciable contribution to the plant CDF.

6. **The Specific Site Characteristics**

AP1000 Chapter 49, "Offsite Dose Risk Quantification," is based on an EPRI report (Reference 11) to establish the specific site characteristics for AP1000. Reference 11 Annex B, "ALWR Reference Site," establishes a conservative reference site to represent the consequences of most potential sites with respect to exposure at the site boundary. This reference site was based on the characteristics of 91 U.S. reactor sites that are tabulated in the NRC document, "Technical Guidance for Siting Criteria Development," (NUREG CR-2239) (Reference 12). Annex B provides a summary of the meteorological data to be used in calculating off-site dose. The following information has been obtained from Reference 12.

Base Case Used for Siting Criteria Development

- A standard 1120 MWe PWR

Chosen because many reactors operating in December 1982 and most under construction at that time were about this size.

- AN SST1 release

For the purpose of decision-making in such areas as siting and emergency response, NRC defined a set of five siting source terms (SST1-5) to represent the five accident groups. SST1 is as follows:

- Severe core damage
- Loss of all installed safety features
- Severe direct breach of containment

Based on, at that time (December 1982) the number of PRAs available, NRC suggested that a representative probability for SST1 would be 1E-05. SST1 was then normalized to 100 and comparisons and sensitivities were made to the other four SSTs for conditional mean consequences of:

- Mean early fatalities – Indian Point 710-1300*
  The maximum distance, to which early fatalities occurred for an SST1 release ranged from 13 to 25 miles, depending on meteorology, and 18 miles for New York meteorology. Improbable events with conditional probabilities of $\leq$ E-03 caused by adverse weather, e.g., rainout of the radioactive plume onto a population center.
- Mean early injuries – Indian Point 2400-14000*

- Mean latent cancer fatalities – Indian Point 7600-9300*
- Mean thyroid modules
- Mean interdicted land area – Indian Point 72-140 sq. miles*

* All #s based on Complementary Cumulative Distribution Functions (CCDFs) generated with meteorological data from 29 National Weather Service Stations

- New York City meteorology

  Observed rainfall for New York is 697 hours and annual rain of 49 inches

- Indian Point wind rose and population

- Summary evacuation

Summary of Consequence Distances (Miles)

Source	Consequence	Conditional Probability Level		
		Mean	99%	Calc Max
SST1	Early Fatalities	<5	≤15	<25
	Early Injuries	~10	~30	≥50
	Land Interdiction	~20	>50	>50
	PAGs	≥50	>50	>50

Mean distances are the average of the probability distributions of distance.
99% distances refer to those beyond which a consequence or dose is calculated to occur in 1 in 100 accidents.
Calculated maxima represent the largest distances calculated
PAG is defined as the "projected" dose to an individual in the general public that warrants the initiation of emergency protective actions. PAGs range from 1 to 5 rems for whole body exposure and from 5 to 25 rems for projected dose to the thyroid.

Sensitivity of Fatal, Injury, and Interdiction Distances to Release Magnitude

Source (mi)	Fatal Distance (mi)			Injury Distance (mi)			Interdiction Distance		
SST1	Mean	99%	Peak	Mean	99%	Peak	Mean	99%	Peak
1	3.9	1.2	18	11	35	50	19	55	85
½	2.5	10	18	7.0	20	25	14	45	50
1/10	0.9	2.2	5.0	2.8	10	18	5.5	18	25
1/20	0.5	2.0	2.0	1.9	7.0	10	3.6	12	18
1/100	0	1.0	1.0	0.9	4.0	5.0	1.1	10	10

Peak result is that obtained for the most favorable weather conditions sampled.

7. Value of Eliminating Risk

The cost benefit methodology of NUREG/BR-0184 (1997) is used to calculate the maximum attainable benefit. This includes replacement power costs.

For this purpose, the change in the CDF frequency (delta-F) is assumed to be equal to the sum of CDF frequencies from internal, external, and shutdown events that are already evaluated:

delta F = 4 E-07/year.

This is bounding, used to calculate the maximum attainable benefit. In practice, there is no design alternative, or SAMDA strategy, whose implementation would reduce the plant CDF to zero (or to an infinitesimally small frequency).

Table 49-10 Rev. 0 has the higher average person-rem risk (compared to the dose calculations submitted in Table 49-10 Rev.1), it is used to calculate the person-rem exposure:

Dose = 60000 person-rem (0.0144 / 2.41E-07, from Table 49-10).

It is assumed that this dose is applicable to all events (internal, external, at-power, shutdown). Thus, the consequences (dose and other) from all events is included in the calculations.

The following cost categories are investigated:

C1	Public Health (Accident)		5.7.1	5.7.1.3	W(pha)
C2	Public Health (Routine)		5.7.2	5.7.2	V(phr)
C3	Occupational Health (Accident)	Sum of C4 and C5	5.7.3	5.7.3	V(oha)
C4		Accident Related Exposure - ID		5.7.3.3	W(io)
C5		LT Doses		5.7.3.3	W(lto)
C6	Occupational Health (Routine)		5.7.4	5.7.4	V(ohr)
C7	Offsite property		5.7.5	5.7.5	V(fp)
C8	Onsite property	Sum of C9, C10 and C11	5.7.6	5.7.6	V(op)
C9		Cleanup and decon		5.7.6.1	U(cd)
C10		LT replacement power		5.7.6.2	U(rp)
C11		repair and refurbishment		5.7.6.3	

The calculations are given in Table 7-1.

The present-dollar value equivalent for severe accidents at one unit of AP1000 is the sum of the offsite exposure costs, offsite economic costs, onsite exposure costs, and onsite economic costs. The present-day value (at 7% discount rate) of eliminating all plant CDF (maximum attainable benefit) is calculated to be $16,000, which is a very small dollar value. Thus, any mitigating system or a SAMDA strategy/alternative that reduces the plant risk by a fraction of the total plant CDF must cost less than $16,000 to be cost-effective. This result maintains the same conclusions as stated previously in the RAI for the cost-effectiveness of the SAMDA alternatives.

Another calculation of the maximum attainable benefit is made with the discount rate of 3% (Table 7-2). The resulting value is $33,000, which is still very small to justify any appreciable investment.

Even if a very conservative multiplicative error factor of 10 were used, the maximum attainable benefit would be limited to a cost below $160,000.

The following table summarizes the results of the base case and 4 sensitivity cases:

Base Case	7% Discount rate	$16,000
SC-1	3 % Discount rate	$33,000
SC-2	High Dose	$20,000
SC-3	Realistic delta-F	$8,000
SC-4	Twice the CDF	$31,000
SC-5	Mult. EF of 10	$157,000

In all cases, the values are strongly affected (increased) because of the replacement power cost. This is an inappropriate bias for public decision making, since it does not relate to public safety and it is not even a direct cost to the public since the costs are to the utility, and their impact on the electricity rates for the public is unpredictable.

The first sensitivity case is already discussed above. In the second sensitivity case (Table 7-3), the dose values are increased (10 times for external, NUREG high-estimates for occupational health). The third sensitivity analysis acknowledges that the delta-F realistically can not be equal to the total plant CDF; a factor of 0.5 in introduced.

Sensitivity case 4 examines the case where the CDF value (thus the delta-F) is increased by a factor of 2. Finally, sensitivity analysis case 5 looks at what happens if a multiplicative error factor of 10 is applied to the base case. In all cases, the benefits range from very small to modest.

Table 7-1 AP1000 Calculation of Value of Risk Reduction

Delta F = total CDF =	4.00E-07	events/year
r =	0.07	
ti =	5.0	years
tf =	40	years
m =	10	years
C =	9.198E+00	(exp(-rti) - exp(-rtf))/r

	delta F * PV	Present Value of Cost Over Lifetime			NUREG/BR - 0184		
C1	$442	1.10E+09	Public Health (Accident)		5.7.1	5.7.1.3	W(pha)
C2	$0		Public Health (Routine)		5.7.2	5.7.2	V(phr)
C3			Occupational Health (Accident)	Sum of C4 and C5)	5.7.3	5.7.3	V(oha)
C4	$24	6.07E+07		Accident Related Exposure - ID		5.7.3.3	W(io)
C5	$150	3.75E+08		LT Doses		5.7.3.3	W(lto)
C6	$0		Occupational Health (Routine)		5.7.4	5.7.4	V(ohr)
C7	$905	2.26E+09	Offsite property		5.7.5	5.7.5	V(fp)
C8			Onsite property	Sum of C9, C10 and C11	5.7.6	5.7.6	V(op)
C9	$5,789	1.45E+10		Cleanup and decon		5.7.6.1	U(cd)
C10	$8,376	2.09E+10		LT replacement power		5.7.62	U(rp)
C11	$0			repair and refurbishment		5.7.6.3	
Sum =	$15,686	3.92E+10					

Calculations:

Calc 1		Population Dose (average) =	6.000E+04	person-rem
		R = Monetary Equivalent of unit dose =	2.000E+03	dollars/person-rem
		Z(PHA) =	1.200E+08	dollars
		W(PHA) = C * Z(PHA) =	1.104E+09	
	C1 =	W(PHA) * CDF =	4.415E+02	
Calc 2	C2 =	0		
Calc 3	C3 =	C4 + C5 =	1.745E+02	
Calc 4		D(IO) =	3.300E+03	person-rem
		Z(IO) =	6.600E+06	
		C =	9.198E+00	
		W(IO) = C * Z(IO) =	6.071E+07	
	C4 =	W(IO) * CDF =	2.428E+01	

Calc 5		D(LTO) =		2.000E+04	person-rem
		Z(LTO) =	D(LTO)* CDF * R =	1.600E+01	
	C5	W(LTO) = (Z(LTO)/m*r^2)(1-exp(-r(tf-ti)))(1-exp(-rm)		1.502E+02	
Calc 6	C6 =	0			
Calc 7		B = (average of Table 5.6) =		2.460E+08	
		B* CDF =	Bt	9.840E+01	
	C7 =	D = C * Bt		9.051E+02	
Calc 8		=C9 + C10		1.417E+04	
Calc 9		C(CD) =		1.500E+09	
		PV(CD) =	(C(CD)/mr)*(1-exp(-rm))	1.079E+09	
		U(CD) = (PV(CD)/r)*(1-exp(-rtf))		1.447E+10	
	C9	U(CD) * CDF =		5.789E+03	

Calc 10		PV(RP) =	1.2E+08 / r * (1-exp(-rtf))^2	1.512E+09
		U(RP) = (PV(RP)/r) (1-exp(-rtf))^2		1.905E+10
		U(RP)*1000/910		2.094E+10
	C10	U(RP)*1000/910 * CDF		8.376E+03
Calc 11	C11	0		

Table 7-2 AP1000 Calculation of Value of Risk Reduction - 3% discount rate

DeltaF = total CDF =	4.00E-07	events/year
r =	0.03	
ti =	5.0	years
tf =	40	years
m =	10	years
C =	1.865E+01	(exp(-rti) - exp(-rtf))/r

	delta F * PV	Present Value of Cost Over Lifetime			NUREG/BR - 0184		
C1	$895	2.24E+09	Public Health (Accident)		5.7.1	5.7.1.3	W(pha)
C2	$0		Public Health (Routine)		5.7.2	5.7.2	V(phr)
C3			Occupational Health (Accident)	Sum of C4 and C5)	5.7.3	5.7.3	V(oha)
C4	$49	1.23E+08		Accident Related Exposure - ID		5.7.3.3	W(io)
C5	$300	7.49E+08		LT Doses		5.7.3.3	W(lto)
C6	$0		Occupational Health (Routine)		5.7.4	5.7.4	V(ohr)
C7	$1,835	4.59E+09	Offsite property		5.7.5	5.7.5	V(fp)
C8			Onsite property	Sum of C9, C10 and C11	5.7.6	5.7.6	V(op)
C9	$12,075	3.02E+10		Cleanup and decon		5.7.6.1	U(cd)
C10	$17,788	4.45E+10		LT replacement power		5.7.62	U(rp)
C11	$0			repair and refurbishment		5.7.6.3	
Sum =	$32,941	8.24E+10					

Calculations:

Calc 1		Population Dose (average) =	6.000E+04	person-rem
		R = Monetary Equivalent of unit dose =	2.000E+03	dollars/person-rem
		Z(PHA) =	1.200E+08	dollars
		W(PHA) = C * Z(PHA) =	2.238E+09	
	C1 =	W(PHA) * CDF =	8.952E+02	
Calc 2	C2 =	0		
Calc 3	C3 =	C4 + C5 =	3.488E+02	
Calc 4		D(IO) =	3.300E+03	person-rem
		Z(IO) =	6.600E+06	
		C =	1.865E+01	
		W(IO) = C * Z(IO) =	1.231E+08	
	C4 =	W(IO) * CDF =	4.924E+01	

Calc 5		D(LTO) =		2.000E+04	person-rem
		Z(LTO) =	D(LTO)* CDF * R =	1.600E+01	
	C5	W(LTO) = (Z(LTO)/m*r^2)(1-exp(-r(tf-ti)))(1-exp(-rm)		2.995E+02	
Calc 6	C6 =	0			
Calc 7		B = (average of Table 5.6) =		2.460E+08	
		B* CDF =	Bt	9.840E+01	
	C7 =	D = C * Bt		1.835E+03	
Calc 8		=C9 + C10		2.986E+04	
Calc 9		C(CD) =		1.500E+09	
		PV(CD) =	(C(CD)/mr)*(1-exp(-rm))	1.296E+09	
		U(CD) = (PV(CD)/r)*(1-exp(-rtf))		3.019E+10	
	C9	U(CD) * CDF =		1.207E+04	

Calc 10		PV(RP) = 1.2E+08 / r * (1-exp(-rtf))^2	1.953E+09
		Correction factor 1.4/1.1 * PV(RP)	2.486E+09
		U(RP) = (PV(RP)/r) (1-exp(-rtf))^2	4.047E+10
		U(RP)*1000/910	4.447E+10
	C10	U(RP)*1000/910 * CDF	1.779E+04
Calc 11	C11	0	

Table 7-3 AP1000 Calculation of Value of Risk Reduction - High dose estimates

(NUREG high estimates for D(IO) and D(LTO), 10 times the population dose for CDF)

DeltaF = total CDF =	4.00E-07	events/year
r =	0.07	
ti =	5.0	years
tf =	40	years
m =	10	years
C =	9.198E+00	(exp(-rti) - exp(-rtf))/r

	delta F * PV	Present Value of Cost Over Lifetime			NUREG/BR - 0184		
C1	$4,415	1.10E+10	Public Health (Accident)		5.7.1	5.7.1.3	W(pha)
C2	$0		Public Health (Routine)		5.7.2	5.7.2	V(phr)
C3			Occupational Health (Accident)	Sum of C4 and C5)	5.7.3	5.7.3	V(oha)
C4	$103	2.58E+08		Accident Related Exposure - ID		5.7.3.3	W(io)
C5	$225	5.63E+08		LT Doses		5.7.3.3	W(lto)
C6	$0		Occupational Health (Routine)		5.7.4	5.7.4	V(ohr)
C7	$905	2.26E+09	Offsite property		5.7.5	5.7.5	V(fp)
C8			Onsite property	Sum of C9, C10 and C11	5.7.6	5.7.6	V(op)
C9	$5,789	1.45E+10		Cleanup and decon		5.7.6.1	U(cd)
C10	$8,376	2.09E+10		LT replacement power		5.7.62	U(rp)
C11	$0			repair and refurbishment		5.7.6.3	
Sum =	$19,814	4.95E+10					

Calculations:

Calc 1		Population Dose (average) =	6.000E+05	person-rem
		R = Monetary Equivalent of unit dose =	2.000E+03	dollars/person-rem
		Z(PHA) =	1.200E+09	dollars
		W(PHA) = C * Z(PHA) =	1.104E+10	
	C1 =	W(PHA) * CDF =	4.415E+03	
Calc 2	C2 =	0		
Calc 3	C3 =	C4 + C5 =	3.283E+02	
Calc 4		D(IO) =	1.400E+04	person-rem
		Z(IO) =	2.800E+07	
		C =	9.198E+00	
		W(IO) = C * Z(IO) =	2.576E+08	
	C4 =	W(IO) * CDF =	1.030E+02	

Calc 5		D(LTO) =	3.000E+04	person-rem
		Z(LTO) = D(LTO)* CDF * R =	2.400E+01	
	C5	W(LTO) = (Z(LTO)/m*r^2)(1-exp(-r(tf-ti)))(1-exp(-rm)	2.253E+02	
Calc 6	C6 =	0		
Calc 7		B = (average of Table 5.6) =	2.460E+08	
		B* CDF = Bt	9.840E+01	
	C7 =	D = C * Bt	9.051E+02	
Calc 8		=C9 + C10	1.417E+04	
Calc 9		C(CD) =	1.500E+09	
		PV(CD) = (C(CD)/mr)*(1-exp(-rm))	1.079E+09	
		U(CD) = (PV(CD)/r)*(1-exp(-rtf))	1.447E+10	
	C9	U(CD) * CDF =	5.789E+03	

Calc 10		PV(RP) =	1.2E+08 / r * (1-exp(-rtf))^2	1.512E+09
		U(RP) = (PV(RP)/r) (1-exp(-rtf))^2		1.905E+10
		U(RP)*1000/910		2.094E+10
	C10	U(RP)*1000/910 * CDF		8.376E+03
Calc 11	C11	0		

8. Evaluation of Potential Improvements

We have already estimated value of eliminating AP1000 total risk as $16,000, as discussed in Section 7 above. This value is an upper bound for any single engineered design alternative, which would actually reduce CDF and/or LRF a fraction of the values assumed in the base case for calculating the $16,000 value. Moreover, only three percent of the $16,000 comes from reduction of man-rem exposure. Thus, any design alternative that does not reduce CDF considerably, even if it does reduce the man-rem exposure, would not be cost beneficial.

For AP600, 14 design alternatives were discussed in the SAMDA section (Appendix 1B) and were found to be not cost effective. One of these alternatives is actually implemented in the AP1000 design (diverse containment recirculation squib valves). The costs associated with the remaining 13 design alternatives are provided in Table 8.1. Note that only one design alternative, # 3, namely self-actuating containment isolation valves has a cost near $16,000; the remaining alternatives are at least an order of magnitude more costly than $16,000. Thus, only design alternative #3 needs to be further discussed.

Table 8-1 Design Alternatives for SAMDA

#	Design Alternative	Cost
1	Upgrade CVS for Small LOCA	1,500,000
2	Containment Filtered Vent	5,000,000
3	Self-Actuating Containment Isolation Valves	33,000
4	Safety Grade Passive Containment Spray	3,900,000
6	SG Shell Side Heat Removal	1,300,000
7	SG Relief Flow to IRWST	620,000
8	Increased SG Pressure Capability	8,200,000
9	Secondary Containment Ventilation with Filtration	2,200,000
10	Diverse IRWST Injection Valves	570,000
11	Diverse Containment Recirculation Valves	Already Implemented
12	Ex-Vessel Core Catcher	1,660,000
13	High Pressure Containment Design	50,000,000
14	More Reliable DAS	470,000

Self-Actuating Containment Isolation Valves

This SAMDA consists of improved containment isolation provisions on all normally open containment penetrations. The category of "normally open" is limited to normally open pathways to the environment during power and shutdown conditions, excluding closed systems inside and outside the containment such as RNS and component cooling. The design alternative would be to add a self-actuating valve or

enhance the existing inside containment isolation valve to provide for self-actuation in the event that containment conditions are indicative of a severe accident. Conceptually, the design would either be an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve, thus providing the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures, which occur at the time of the accident such as in cases of vessel rupture or anticipated transients without scram (ATWS). This design alternative provides almost no benefit in reducing plant CDF.

Generously assuming that this design alternative will eliminate CI release totally, the frequency of CI is 1.33E-09/yr for internal events at power and its population dose is 2E+06 person rem. The benefit of this design alternative is about $100 (including a factor of 2 to allow for shutdown and external events contribution). Even this cheapest design alternative does not meet the benefit/cost ratio of 1.

Other New design Changes:

Other design changes, as discussed in Section 5, are already incorporated into AP1000. There is no cost/benefit analysis available for those changes already incorporated.

9. Results

Due to existing low risk of the AP1000 plant, none of the design alternatives described in Appendix A meet an acceptable benefit to cost ratio of 1 or greater.

Several of the design alternatives evaluated in other SAMDA analyses are included in the current AP1000 design. These design features include:

RCS depressurization system
Passive residual heat removal system located inside containment
Cavity flooding system
Passive containment cooling system
Hydrogen igniters in a large-dry containment
Diverse actuation system
Canned motor RCPs
Interfacing system with high design pressure

As the AP1000 plant core damage frequency is lower than for existing plants, the benefits of additional design alternatives are very small. The fifteen SAMDAs analyzed provided little or no benefit to the AP1000 design.

Assuming a hypothetical design alternative was developed which provides a 100 percent reduction in overall plant risk, the capital benefit only amounts to $16,000.

10. References

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3. "AP600 Probabilistic Risk Assessment," Westinghouse Electric Corporation and ENEL, Revision 8, September 1996.

4. "Supplement to the Final Environmental Statement - Limerick Generating Station, Units 1 and 2," Docket Nos. 50-352/353, August 1989.

5. "Supplement to the Final Environmental Statement - Comanche Peak Steam Electric Station, Units 1 and 2," Docket Nos. 50-445/446, October 1989.

6. "System 80+ Design Alternatives Report," Docket No. 52-002, April 1992.

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11. EPRI Advanced Light Water Reactor Utility Requirements Document Volume III Annex B "ALWR Reference Site," Revisions 5 & 6, December 1993.

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APPENDIX A

Design Alternatives

This subsection describes each SAMDA and the benefit expected due to the modification. In the evaluation of the risk reduction benefit, each SAMDA is assumed to operate perfectly with 100 percent efficiency, without failure of supporting systems. A perfect SAMDA reduces the frequency of accident sequences which it addresses to zero. This is conservative as it maximizes the benefit of each design alternative. The SAMDA will reduce the risk by lowering the frequency, attenuating the release, or both. The benefit will be described in terms of the accident sequences and dose which are affected by the SAMDAs, as well as the overall risk reduction. Note that for the purposes of these evaluations, increases to release category IC are not factored into the risk benefit calculations. The IC dose is sufficiently small that changes to the IC total frequency do not result in an appreciable change to overall results. This is also a conservative representation since this maximizes the risk reduction.

Since AP1000 alternatives are the same for AP1000 as for AP600, specific AP1000 risk reduction factor calculations were not performed for AP1000. To recognize the effect of the differences in release frequencies between AP600 and AP1000, they were compared. The largest difference in release category frequency between AP600 and AP1000 is for CFI, which is 14.5 times larger in AP1000 than for AP600. For conservatism, each of the AP600 SAMDA risk reduction factors was multiplied by 15 and applied to AP1000.

1 Upgrade the CVS for Small LOCAs

The chemical, volume, and control system (CVS) is currently capable of maintaining the reactor coolant system (RCS) inventory to a level in which the core remains covered in the event of a very small (≤ 3/8" diameter break) loss of coolant accident (LOCAs). This SAMDA involves providing in-containment refueling water storage tank (IRWST) / containment recirculation connections to the CVS and adding a second line from the CVS makeup pumps to the RCS in order to be able to use the system to keep the core covered during small and intermediate LOCAs.

A perfect, upgraded CVS system is assumed to prevent core damage in the RCS leak, passive RHR heat exchanger tube ruptures, small LOCA, and intermediate LOCA release categories. The CVS is assumed to have perfect support systems (power supply, component cooling) and to work in all situations regardless of the common cause failures of other systems.

2 **Filtered Vent**

This SAMDA consists of placing a filtered containment vent and all associated piping and penetrations into the AP1000 containment design. The filtered vent could be used to vent the containment to prevent catastrophic overpressure failure, and also provides filtering capability for source term release. With respect to the AP1000 PRA, the possible scenario in which the filtered vent could result in risk reduction would be late containment overpressure failures (release category CFL). Other containment overpressure failures occur due to dynamic severe accident phenomena, such as hydrogen burn, steam explosion, etc. The late containment failures for AP1000 are failures of the passive containment cooling system (PCS). Analyses have indicated that for scenarios with PCS failure, air cooling may limit the containment pressure to less than the ultimate pressure. However, for the purposes of the Level 2 PRA, failure of PCS is assumed to result in containment failure based on an adiabatic heatup. To conservatively consider the risk reduction of a filtered vent, the use of a filtered vent to preclude a late containment failure will be evaluated. A decontamination factor (DF) of 1000 will conservatively be assumed for each PRA Level 1 accident classification, even though it is realized that the dose due to noble gases will not be impacted by the filtered vent since 100% of the noble gas fission products will still be released. Therefore, the risk reduction is equal to the decontamination factor assumed, since the PRA Level 1 accident classification frequencies do not change.

3 **Self-Actuating Containment Isolation Valves**

This SAMDA consists of improved containment isolation provisions on all normally open containment penetrations. The category of "normally open" is limited to normally open pathways to the environment during power and shutdown conditions, excluding closed systems inside and outside the containment such as RNS and component cooling. The design alternative would be to add a self-actuating valve or enhance the existing inside containment isolation valve to provide for self-actuation in the event that containment conditions are indicative of a severe accident. Conceptually, the design would either be an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve, thus providing the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures, which occur at the time of the accident such as in cases of vessel rupture or anticipated transients without scram (ATWS).

4 **Passive Containment Sprays**

This SAMDA involves adding a passive safety-related spray system and all associated piping and support systems to the AP1000 containment. A passive containment spray system could result in risk benefits in the following ways:

¥ scrubbing of fission products, primarily for CI failures,

¥ assuming appropriate timing, containment spray could be used as an alternate means for flooding the reactor vessel (in-vessel retention) and for debris quenching should vessel failure occur,

¥ containment spray could also be used to control containment pressure for cases in which PCS has failed.

In order to envelop these potential risk benefits, the risk reduction evaluation will assume that containment sprays are perfectly effective for each of these benefits, with the exception of fission product scrubbing for containment bypass. Thus the risk reduction can be conservatively estimated by assuming all release categories except BP are eliminated.

5 **Active High Pressure Safety Injection System**

This SAMDA consists of adding a safety-related active high pressure safety injection (HPSI) pump and all associated piping and support systems to the AP1000 design. A perfect high pressure safety injection system is assumed to prevent core melt for all events but excessive LOCA and ATWS. Therefore, to estimate the risk reduction, only the contributions to each release category of Level 1 accident classes 3C (vessel rupture) and 3A (ATWS) need be considered. The averted risk is shown in Table 8-1. This SAMDA would completely change the design approach from a plant with passive safety systems to a plant with passive plus active safety-related systems and is not consistent with design objectives.

6 **Steam Generator Shell-Side Heat Removal System**

This SAMDA consists of providing a passive safety-related heat removal system to the secondary side of the steam generators. The system would provide closed loop cooling of the secondary using natural circulation and stored water cooling, thus preventing a loss of primary heat sink in the event of a loss of startup feedwater and passive RHR heat exchanger. A perfect secondary heat removal system would eliminate transients from each of the release categories. In order to evaluate the benefit of this SAMDA, the frequencies of all the transient sequences

is subtracted from the overall frequency of each of the release categories and the risk is recalculated. The total risk averted is shown in Table 8-1.

7 **Direct Steam Generator Relief Flow to the IRWST**

This SAMDA consists of providing all the piping and valves required for redirecting the flow from the steam generator safety and relief valves to the in-containment refueling water storage tank (IRWST). An alternate, lower cost option of this SAMDA consists of redirecting only the first stage safety valve to the IRWST. This system would prevent or reduce fission product release from bypassing the containment in the event of a steam generator tube rupture (SGTR) event. In order to evaluate the benefit from this SAMDA (both options), this design change is assumed to eliminate the BP release category.

8 **Increased Steam Generator Pressure Capability**

This SAMDA consists of increasing the design pressure of the steam generator secondary side and safety valve set point to the degree that a steam generator tube rupture will not cause the secondary system safety valve to open. The design pressure would have to be increased sufficiently such that the combined heat capacity of the secondary system inventory and the PRHR system could reduce the RCS temperature below T_{sat} for the secondary design pressure. Although specific analysis would have to be performed, it is estimated that the design pressure would have to be increased several hundred psi. This design would also prevent the release of fission products which bypasses the containment via the SGTR.

9 **Secondary Containment Filtered Ventilation**

This SAMDA consists of providing the middle and lower annulus (below the 135′ 3″ elevation) of the secondary concrete containment with a passive annulus filter system to for filtration of elevated releases. The passive filter system is operated by drawing a partial vacuum on the middle annulus through charcoal and HEPA filters. The partial vacuum is drawn by means of an eductor with motive flow from compressed gas tanks. The secondary containment would then reduce particulate fission product release from any failed containment penetrations (containment isolation failure). In order to evaluate the benefit from such a system, this design change is assumed to eliminate the CI release category.

10 **Diverse IRWST Injection Valves**

This SAMDA consists of changing the in-containment refueling water storage tank (IRWST) injection valve designs so that two of the four lines use diverse valves.

Each of the four lines is currently isolated by a squib valve in series with a check valve. In order to provide diversity, the valves in two of the lines will be provided by a different vendor. For the check valves, alternate vendors are available. However, it is questionable if check valves of different vendors would be sufficiently different to be considered diverse unless the type of check valve was changed from the current swing disk check to another type. The swing disk type is the preferred type for this application and other types are considered to be less reliable. Squib valves are specialized valve designs for which there are few vendors. A vendor may not be willing to design, qualify, and build a reasonable squib valve design for this AP1000 application considering that they would only supply two valves per plant. As a result, this SAMDA is not really practicable because of the uncertainty in availability of a second squib valve design/vendor and because of the uncertainty in the reliability of another check valve type. However, the cost estimate for this SAMDA assumes that a second squib valve vendor exists and that vendor only provides the two diverse IRWST squib valves. The cost impact does not include the additional first time engineering and qualification testing that will be incurred by the second vendor. Those costs are expected to be more than a million dollars.

This change will reduce the frequency of core melt by eliminating the common cause failure of the IRWST injection. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of IRWST injection are assumed to be averted. Core damage sequences resulting from a failure of IRWST injection correspond to PRA Level 1 accident classification 3BE; thus, release category 3BE is eliminated.

11 Diverse Containment Recirculation Valves

This SAMDA consists of changing the containment recirculation valve designs so that two out of the four lines use diverse valves. Each of the four lines currently contains a squib valve; two of the lines contain check valves and the other two contain motor-operated valves. In order to provide diversity, the squib valves in two lines will be made diverse by supplying them from a different vendor. This change will reduce the frequency of core melt by eliminating the common cause failure of the containment recirculation. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of containment recirculation are assumed to be averted. Core damage sequences resulting from failure of containment recirculation correspond to PRA Level 1 accident classification 3BL; thus, release category 3BL is eliminated.

12 **Ex-Vessel Core Catcher**

This SAMDA consists of designing a structure in the containment cavity or using a special concrete or coating which will inhibit core-concrete interaction (CCI), even if the debris bed dries out. A perfect core catcher would prevent CCI for all cases. However, the AP1000 incorporates a wet cavity design in which ex-vessel cooling is used to maintain the core debris in the vessel thus preventing ex-vessel phenomena, such as CCI. Consequently, containment failure due to CCI is not considered in detail for the AP1000 Level 2 PRA. For cases in which reactor vessel flooding is failed, it is assumed that containment failure occurs due to ex-vessel steam explosion or CCI. This containment failure is assumed to be an early containment failure, CFE, (due to ex-vessel steam explosion) even though CCI and basemat melt-through would be a late containment failure. To conservatively estimate the risk reduction of an ex-vessel core catcher, this design change is assumed to eliminate the CFE release category.

13 **High Pressure Containment Design**

This SAMDA design consists of using the massive high-pressure containment design in which the design pressure of the containment is approximately 300 psi (20 bar) for the AP1000 containment. The massive containment design has a passive containment cooling feature much like the AP1000 containment. The high design pressure is considered only for prevention of containment failures due to severe accident phenomena such as steam explosions and hydrogen detonation. A perfect high pressure containment design would reduce the probability of containment failures, but would have no reduction of the frequency or magnitude of the release from an unisolated containment (containment isolation failure or containment bypass). To estimate the risk reduction of a high-pressure containment design, this design is assumed to eliminate the CFE, CFI and CFL release categories.

14 **Increase Reliability of Diverse Actuation System**

This SAMDA design consists of improving the reliability of the diverse actuation system (DAS) which actuates engineered safety features and allows the operator to monitor the plant status. The design change would add a third instrumentation and control cabinet and a third set of diverse actuation system instruments to allow the use of 2-out-of-3 logic instead of 2-out-of-2 logic. Other changes, such as adding another set of batteries, have not been included in the cost estimates. A perfectly reliable DAS system would reduce the frequency of the release categories by the cumulative frequencies of all sequences in which DAS failure leads to core damage. In order to evaluate the benefit from the DAS system

upgrade, a Level 1 sensitivity analysis assuming perfect reliability of DAS was completed.

15 **Locate Normal Residual Heat Removal Inside Containment**

This SAMDA consists of placing the entire normal residual heat removal system (RNS) and piping inside the containment pressure boundary. Locating the RNS inside the containment would prevent containment bypass due to interfacing system LOCAs (ISLOCA) of the residual heat removal (RHR) system. In past probabilistic risk assessments of current generation nuclear power plants, the ISLOCA is the leading contributor of plant risk because of large offsite consequences. A failure of the valves which isolate the low pressure RHR system from the high pressure RCS causes the RHR system to overpressurize and fail, releasing RCS coolant outside the containment where it cannot be recovered for recirculation cooling of the core. The result is core damage and the direct release of fission products outside the containment.

In the AP1000, the RNS is designed with a higher design pressure than the systems in current pressurized water reactors, and an additional isolation valve is provided in the design. In the probabilistic risk assessment, no ISLOCAs contribute significantly to the core damage frequency of the AP1000 (Reference 2, Chapter 33). Therefore, relocating the RNS of the AP1000 inside containment will provide virtually no risk reduction benefit and will not be investigated further in terms of cost.

APPENDIX B

The release categories for the AP1000 are defined as follows:

IC - intact containment. Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.

CFE - containment failure early. Fission-product release through a containment failure caused by severe accident phenomenon occurring after the onset of core damage but prior to core relocation.

CFI - containment failure intermediate. Fission-product release through a containment failure caused by severe accident phenomenon occurring after core relocation but before 24 hours.

CFL - containment failure late. Fission-product release through a containment failure caused by severe accident phenomenon occurring after 24 hours.

CI - containment isolation failure. Fission-product release through a failure of the system or valves that close the penetrations between the containment and the environment. Containment failure occurs prior to onset of core damage.

BP - containment bypass. Fission products are released directly from the Reactor Coolant System to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.

The following subsections present a brief description of the AP1000 release categories.

The release category frequencies are shown in Table B-1.

1 Release Category IC - Intact Containment

If the containment integrity is maintained throughout the accident, then the release of radiation from the containment is due to nominal leakage and is expected to be within the design basis of the containment. This is the "no failure" containment failure mode and is termed intact containment. The main location for fission-product leakage from the containment is penetration leakage into the auxiliary building where significant deposition of aerosol fission products may occur.

2 **Release Category CFE - Early Containment Failure**

Early containment failure is defined as failure that occurs in the time frame between the onset of core damage and the end of core relocation. During the core melt and relocation process, several dynamic phenomena can be postulated to result in rapid pressurization of the containment to the point of failure. The combustion of hydrogen generated in-vessel, steam explosions, and reactor vessel failure from high pressure are major phenomena postulated to have the potential to fail the containment. If the containment fails during or soon after the time when the fuel is overheating and starting to melt, the potential for attenuation of the fission-product release diminishes because of short fission-product residence time in the containment. The fission products released to the containment prior to the containment failure are discharged at high pressure to the environment as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Containment failures postulated within the time of core relocation are binned into release category CFE.

3 **Release Category CFI - Intermediate Containment Failure**

Intermediate containment failure is defined as failure that occurs in the time frame between the end of core relocation and 24 hours after core damage. After the end of the in-vessel fission-product release, the airborne aerosol fission products in the containment have several hours for deposition to attenuate the source term. The global combustion of hydrogen generated in-vessel from a random ignition prior to 24 hours can be postulated to fail the containment. The fission products in the containment atmosphere are discharged at high pressure to the environment as the containment blows down. Containment failures postulated within 24 hours of the onset of core damage are binned into release category CFI.

4 **Release Category CFL - Late Containment Failure**

Late containment failure is defined as containment failure postulated to occur later than 24 hours after the onset of core damage. Since the PRA assumes the dynamic phenomena, such as hydrogen combustion, to occur before 24 hours, this failure mode occurs only from the loss of containment heat removal via failure of the passive containment cooling system. The fission products that are airborne at the time of containment failure will be discharged at high pressure to the environment, as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Accident sequences with failure of containment heat removal are binned in release category CFL.

5 **Release Category CI - Containment Isolation Failure**

A containment isolation failure occurs because of the postulated failure of the system or valves that close the penetrations between the containment and the environment. Containment isolation failure occurs before the onset of core damage. For such a failure, fission-product releases from the reactor coolant system can leak directly from the containment to the environment with diminished potential for attenuation. Most isolation failures occur at a penetration that connects the containment with the auxiliary building. The auxiliary building may provide additional attenuation of aerosol fission-product releases. However, this decontamination is not credited in the containment isolation failure cases. Accident sequences in which the containment does not isolate prior to core damage are binned into release category CI.

6. **Release Category BP - Containment Bypass**

Accident sequences in which fission products are released directly from the reactor coolant system to the environment via the secondary system or other interfacing system bypass the containment. The containment failure occurs before the onset of core damage and is a result of the initiating event or adverse conditions occurring at core uncovery. The fission-product release to the environment begins approximately at the onset of fuel damage, and there is no attenuation of the magnitude of the source term from natural deposition processes beyond that which occurs in the reactor coolant system, in the secondary system, or in the interfacing system. Accident sequences that bypass the containment are binned into release category BP.

TABLE B 1 POPULATION WHOLE BODY EDE DOSE RISK – 24 HOURS					
Release Category	Release Frequency (/reactor year)	Mean Dose (person-sieverts)	Dose (person-REM)	Risk (person-REM/ reactor year)	Percent Contribution to Total Risk
CFI	1.89E-10	7.88E+03	7.88E+05	1.49E-04	1.2%
CFE	7.47E-09	8.51E+03	8.51E+05	6.36E-03	51.3%
IC	2.21E-07	7.19E+00	7.19E+02	1.59E-04	1.3%
BP	1.05E-08	2.91E+03	2.91E+05	3.06E-03	24.7%
CI	1.33E-09	2.01E+04	2.01E+06	2.67E-03	21.6%
CFL	3.45E-13	5.32E+03	5.32E+05	1.84E-07	0.0%
			Total Risk =	1.24E-02	100.0%

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.092 (Response Revision 1)

Question:

What is the philosophy of igniter placement in the upper compartment? The AP600 containment volume is 1.69×10^6 cubic feet (ft^3) as compared to 2.07×10^6 ft^3 for the AP1000. Presumably, the bulk of the increase in volume is in the upper compartment. Given that there is a certain volume of coverage for each igniter in the AP600, how does the AP1000 igniter scheme cover the increase in volume of 380,000 ft^3 with the same number of igniters?

Westinghouse Revision 0 Response:

For AP1000 hydrogen control is provided at three separate levels within the upper compartment. At the 162-176 ft. elevations, 10 igniters are distributed over the area primarily above the major release flow paths including the loop compartments, refueling cavity, pressurizer compartment and above the stairwell from the lower compartment area. The igniters are split between the two power groups. At 228 ft. elevation, an igniter is provided in each quadrant at the mid region of the upper compartment with two igniters on each of the two power groups. At the upper region elevation 257 ft. four additional igniters are located to initiate recombination of hydrogen not ignited at either the source or along its flow path. The four igniters are split between the two power groups.

In the upper compartment the potential for large hydrogen concentrations and detonation is very low since there are large natural circulation currents driven by the heated releases from the postulated break and the cooling action of the containment shell and there are effectively no barriers for confinement of the break discharges. Additionally, there are no compartmentalization with geometries that increase the potential for detonation. The AP1000 design includes igniters, which would burn hydrogen as it is released such that its global concentration in containment would not exceed 10%. A hydrogen concentration of less than 10% in dry air (note: the AP1000 containment has a steam-laden environment) is not detonable.

The number of igniters and igniter placement for the AP600 and AP1000 designs is based on engineering criteria from insights on how the hydrogen is released during a severe accident and how it is expected to behave in containment. The primary criteria for an igniter system is to promote hydrogen burning at as low a concentration as possible, and to the extent possible, to burn hydrogen continuously so that the hydrogen concentration will not build up in the containment. To achieve this goal igniters have been placed in major regions of the containment where hydrogen may be released, through which it could flow, or where it may accumulate.

The release paths and severe accident behavior for the AP600 and AP1000 designs are similar. In addition, the upper containment dome configuration is identical. Therefore, the number of igniters is the same and igniter placement is similar. As discussed above, igniter number and placement for the AP600 and AP1000 designs is defined by providing igniters in the major transport paths and is not dictated by the volume of hydrogen produced. The major transport paths remain the same for AP600 and AP1000, and therefore, the number of igniters provided is the same. The following design criteria has been established and followed for igniter placement (Reference: AP1000 DCD, Table 6.2.4-6 (Sheet 1 of 3)):

- A sufficient number of igniters are placed in the major transport paths (including dominant natural circulation pathways) of hydrogen so that hydrogen can be burned continuously close to the release point. This prevents hydrogen from preferentially accumulating in a certain region of the containment.

- Igniters (minimum of 2) are located in major regions or compartments where hydrogen may be released, through which it may flow, or where it may accumulate.

- It is preferable to ignite a hydrogen-air mixture at the bottom so that upward flame propagation can be promoted at lean hydrogen concentrations. Igniters within each subcompartment are located in the vicinity of, and above, the highest potential release location within the subcompartment.

- In compartments with relatively small openings in the ceiling, the potential may exist for the hydrogen-air mixture to rise and to collect near the ceiling. Therefore, one or more igniters are placed near the ceiling of such compartments. Igniter coverage is provided within the upper 10 percent of the vertical height subcompartments or 10 feet from the ceiling whichever is less. In cases where the highest potential release point is low in the compartment, both this and the previous criteria are considered.

- To the extent possible, igniters are placed away from walls and other large surfaces so that a flame front created by ignition at the bottom of a compartment can travel unimpeded up to the top.

- A sufficient number of igniters are installed in long, narrow compartments (corridors) so that the flame fronts created by the igniters need to travel only a limited distance before they merge. This limits the potential for significant flame acceleration.

- Igniter coverage is provided to control combustion in areas where oxygen rich air may enter into an inerted region with combustible hydrogen levels during an accident scenario.

- Igniters are located above the flood level, if possible. Those which may be flooded have redundant fuses to protect the power supply.

- In locations where the potential hydrogen release location can be defined, i.e. above the IRWST spargers, at IRWST vents, etc igniter coverage is provided as close to the source as feasible.

- Provisions for installation, maintenance, and testing is considered.

NRC Additional Comment:

The Westinghouse design criteria do not address how many igniters should be placed within each AP1000 compartment. The issue of igniter spacing has not been addressed for the AP600 or AP1000 plants. Please provide the technical basis for the numbers and the placement of igniters in AP1000 containment.

Revise response to address the distance spacing criteria.

Westinghouse Additional Response:

The igniter placement criteria for AP1000, which is the same as the AP600, does not explicitly contain a distance spacing criteria. The igniter placement criteria as described above (also included in DCD Section 6.2) specifies that two igniters be placed in each dead-ended compartment, where hydrogen may collect. In addition, igniters are placed in the containment dome as discussed in the original response. Such a criteria is deemed sufficient, because one igniter per dead-ended compartment is sufficient to preclude an unacceptable buildup of hydrogen in any compartment

For AP1000, igniters are placed in the same locations in the compartments below the operating deck as placed in the AP600. The design of the AP1000 below the operating deck is essentially the same as that of the AP600, and the placement of the igniters are such that there are at least two igniters in each dead-ended compartment. The igniters placed in the AP1000 containment dome are also the same as that of the AP600. The additional containment volume provided in the AP1000 (when compared to AP600) is in the upper compartment above the operating deck, and below the containment dome. In this large open volume, there is no significant mechanism for flame acceleration and no additional hydrogen igniters are requred. .

Design Control Document (DCD) Revision:

None

PRA Revision:

None