Westinghouse

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

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

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

Your ref. Docket No. 52-006
Our ref. DCP/NRC1550

February 21, 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 revised 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

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

/Attachments

1. Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1550"
2. Westinghouse Non-Proprietary Response to US Nuclear Regulatory Commission Requests for Additional Information dated February 2003

D063

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DCP/NRC1550

February 21, 2003

Attachment 1

ATTACHMENT 1

Table 1 "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1550"
RAI 210.012, Revision 1
RAI 420.006, Revision 1
RAI 420.007, Revision 1
RAI 420.008, Revision 1
RAI 420.013, Revision 1
RAI 420.045, Revision 1
RAI 440.043, Revision 1
RAI 440.050, Revision 1
RAI 440.128, Revision 1
RAI 440.183, Revision 0
RAI 440.184, Revision 0
RAI 620.043, Revision 1
RAI 650.001, Revision 0
RAI 650.002, Revision 0
RAI 650.003, Revision 0
RAI 650.004, Revision 0
RAI 650.005, Revision 0
RAI 650.006, Revision 0
RAI 720.005, Revision 1

Attachment 2

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.012 (Response Revision 1)

Original Question:

Reference, Volume 6, Section 3.9.2.5, Pg. 3.9-41, last sentence of Section 3.9.2.5.2:

This section describes the analytical methods used to calculate stresses and deflections in the RPV internals due to the combined loads from postulated pipe rupture and the safe shutdown earthquake. The last sentence in this section states the final conclusion that the reactor internals components are within acceptable stress and deflection limits. This significant conclusion is stated without providing, or referencing, any supporting stress and deflection data from the actual analyses (which presumably have been done in order to reach this conclusion).

Please provide a results summary of analytical data, including comparison to appropriate allowable values, which demonstrates that stress, deflection, and stability criteria for the RPV internals design have been met when subjected to the combined effects of the limiting postulated pipe break, and the safe shutdown earthquake.

Follow-On Comment:

Applied loads for AP1000 may be similar to AP600, but due to differences in AP1000 internals geometry, AP1000 stresses may be higher.

Westinghouse Response (Revision 1):

The design of the AP1000 reactor vessel internals is based on the AP600, Westinghouse Standard 3XL, and other previously licensed Westinghouse plants, and is structurally very similar to those designs. The adequacy of the AP1000 core support structures is demonstrated by estimating stress margins for the core support structure components based on a comparison of differences in loads, configurations, and dimensions between a reference plant and the AP1000. The reference plant used for this comparison is the Standard 3XL because of the similarity of the reactor vessel internals configuration and dimensions with the AP1000. Table 1 shows a comparison between the Standard 3XL plant and the AP1000 parameters that could impact the stresses in the core support structures.

Table 1
Westinghouse Standard 3XL Plant/AP1000 Parameter Comparison

Parameter	Standard 3XL	AP1000
Core Barrel Flange		
Outside Diameter, in.	154.03	154.14
Inside diameter, in.	134.05	134.05
Height		
At Vessel Seating Surface, in.	3.5	3.5
Overall – To Weld, in.	14	14
Core Barrel		
Length, in.	323.61	334.80
Inside diameter, in.	134.75	134.75
Outside diameter, in.	137.75	137.75
Lower core support plate thickness, in.	15	15
Lower core support plate flow holes		
Inlet Diameter, in.	1.9	1.9
Outlet Diameter, in.	2.75	2.375
Inlet nozzle ID @ core barrel ID, in. / number of nozzles	36.3 / 3	30.51 / 4
Number of Fuel Assemblies	157	157
Upper Support Columns		
Number	40	42
Outside Diameter, in.	3.5	3.5
Length, in.	83.7	83.7
Outlet nozzle ID @ core barrel ID, in. / number of nozzles	36.6 / 3	43.0 / 2
Upper support structure plate thickness, in.	12	12
Upper support skirt OD, in. / thickness, in.	133.08/2.5	133.08/2.5
Upper support skirt length, in.	23.237	34.3

Service Level D LOCA loads for the Standard 3XL plant included 1 square foot or larger breaks. 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 limiting design basis break to determine the dynamic response of the AP1000 reactor vessel internals is that of a 4" pipe (pressurizer spray line and first stage ADS line). The Standard 3XL plant LOCA loads and stresses resulting from a 1 square foot break were modified where appropriate to account for the small dimensional differences between the Standard 3XL and the AP1000. The resulting loads/stresses were used to evaluate the stress margins for the AP1000 core support structures. A comparison of the loads on the AP600 core support structures resulting from the 1 square foot LOCA with those from a 3" pipe break show that the hydraulic and resulting mechanical loads from the 1 square foot LOCA are approximately an order of magnitude higher. Therefore, the LOCA loads used in the AP1000 evaluation are very conservative.

Preliminary seismic analyses of the AP1000 reactor vessel internals have been completed. The resulting seismic loads were significantly less (by a factor of at least 5) than the corresponding loads resulting from the LOCA. These seismic loads were combined with the conservative LOCA loads from the Standard 3XL plant to evaluate the stress margins in the AP1000 core support structures. For Service Level D conditions the seismic and LOCA loads are combined as the square root of the sum of the square of each load (See AP1000 DCD Table 3.9-5).

Margins to allowable stresses under Level D Conditions were estimated for the AP1000 utilizing the enveloping LOCA loads/stresses from the Standard 3XL plant analysis (modified for dimensional differences between the AP1000 and Standard 3XL), the preliminary AP1000 seismic loads, and the Standard 3XL plant margins to allowable stresses. Table 2 summarizes these margins for the AP1000 core support structures.

Table 2
AP1000 Core Support Structures Preliminary Margins to Allowable Stresses
Service Level D Conditions

Component	Margin (1)	
	P_m (2)	$P_m + P_b$ (3)
Core Barrel		
Flange & Upper Barrel Pin	0.75	0.38
Outlet Nozzles	>1.18	>0.25
Lower Barrel	0.05	0.20
Lower Core Support Plate-Core Barrel Weld	8.5	13.3
Upper Support Column	>0.37	>0.13
Upper Support Structure	>4	0.89
Upper Core Plate Pin	0.24	0.10
Lower Radial Restraint	0.04 (Based on collapse load)	

(1) Margin = (Allowable Stress / Calculated Stress) – 1
(2) Margin in primary membrane stress
(3) Margin in primary membrane plus primary bending stresses

Table 2 shows positive margins to allowable stresses for all AP1000 core support structures, even though the assumed LOCA loads are very conservative. Therefore, the AP1000 core support structures will be within acceptable stress limits for the Service Level D postulated pipe rupture combined with the safe shutdown earthquake.

DCD section 3.9.8.2 provides the commitment that Combined License Applicants referencing the AP1000 design will have available the design specifications and design reports prepared for ASME Section III components. The design reports will include the final stress analyses for the reactor vessel internals.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.006 (Response Revision 1)

Question:

420.6 (DCD 7.1.2.14.2)

Please provide the Commercial Dedication process ITAAC for staff review.

Westinghouse Response:

The Commercial Dedication ITAAC is located in Table 2.5.2-8, Item #13. A copy is provided below.

Table 2.5.2-8 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. The use of commercial grade computer hardware and software items in the PMS is accomplished through a process that specifies requirements for: a) Review of supplier design control, configuration management, problem reporting, and change control. b) Review of product performance. c) Receipt acceptance of the commercial grade item d) Acceptance based on equipment qualification and software validation in the integrated system.	Inspection will be performed of the process defined to use commercial grade components in the application.	A report exists and concludes that the process has requirements for: a) Review of supplier design control, configuration management, problem reporting, and change control. b) Review of product performance. c) Receipt acceptance of the commercial grade item. d) Acceptance based on equipment qualification and software validation in the integrated system.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

Revise response and DCD to include the EPRI topical report as a Tier 2* commitment.

Westinghouse Additional Response:

EPRI TR-106439 will be added to DCD Tier 2 section 7.1.2.14.2, 7.1.7 and the DCD Table 1-1 as shown below.

Design Control Document (DCD) Revision:

Table 1-1
Index of AP1000 Tier 2 Information Requiring NRC Approval for Change

Item	Expiration at First Full Power	Tier 2 Reference
WCAP-13383, "AP600 Instrumentation and Control Hardware & Software Design, Verification & Validation Process Report," Rev 1.	Yes	Chapter 7 Table 1.6-1
WCAP-14605, "Westinghouse Setpoint Methodology for Protection Systems, AP600," Rev 0	Yes	Chapter 7 Table 1.6-1
CE-CES-195, "Software Program Manual for Common Q Systems," Rev 01	Yes	Chapter 7 Table 1.6-1
WCAP-15927, "Design Process for AP1000 Common Q ?Safety Systems," Rev 0	Yes	Chapter 7 Table 1.6-1
Verification and Validation	Yes	7.1.2.14
Design Process	Yes	**7.1.2.14.1**
Commercial Dedication	Yes	**7.1.2.14.2**

Item	Expiration at First Full Power	Tier 2 Reference
TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications"	**Yes**	**7.1.2.14.2** **7.1.7**

7.1.2.14.2 Commercial Dedication

[*WCAP-13383 (Reference 3) and CENPD-396-P (Reference 8) provide for the use of commercial off-the-shelf hardware and software through a commercial dedication process. The commercial dedication process will follow the guidelines outlined in EPRI TR-106439 (Reference 15)*]* Control of the hardware and software during the operational and maintenance phase is the responsibility of the Combined License applicant as described in subsection 13.5.1.

7.1.7 References

[*15. TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," Electric Power Research Institute, October 1996.*]*

***NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.**

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.007 (Response Revision 1)

Question:

420.7 (DCD 7.1.2.5)

Describe the Qualified Data Processing Subsystems (QDPS) in more detail. DCD takes credit on the QDPS in the Defense-in-Depth and Diversity analysis. Describe the relationship between the QDPS and the RTS/ESFAS. Are the QDPS sharing sensors with the RTS or the ESFAS? (Figure 7.1-1 showing some sensors feed into plant protection subsystem directly while some sensors feed into QDPS subsystem). If the QDPS has separate sensors, do they have same qualification requirement as RTS/ESFAS ?

Westinghouse Response:

The Qualified Data Processing Subsystem (QDPS), a subsystem of the Protection and Safety Monitoring System (PMS), provides safety-related display of selected parameters in the control room. Power is provided to the QDPS from the Class 1E dc and UPS system for 72 hours after a loss of all ac power (station blackout). After 72 hours, the ancillary diesel generators provide power for the QDPS. QDPS is a two-train subsystem (Divisions B and C). The PMS, including the QDPS, is diverse from the Diverse Actuation System (DAS). Sensors are not shared between PMS and DAS.

The RTS/ESFAS signals are processed by the Plant Protection Subsystem of PMS. Within PMS, some sensors are shared between the Plant Protection Subsystem and QDPS. Shared sensors are processed first by the QDPS because the QDPS will need this sensor for more than 24 hours following a station blackout. 24-hour batteries power the Plant Protection Subsystem; therefore, the Plant Protection Subsystem can not be used for QDPS functions.

The typical input parameter for RTS/ESFAS is four-way redundant with one sensor for each of the four divisions. If that parameter is also needed by QDPS, the B and C division sensors are processed first by QDPS then sent to the Plant Protection Subsystem. The A and D division sensors are not shared with QDPS and thus are processed directly by the Plant Protection Subsystem. If an RTS/ESFAS parameter is not needed by QDPS, it is processed directly by the Plant Protection Subsystem in all four divisions.

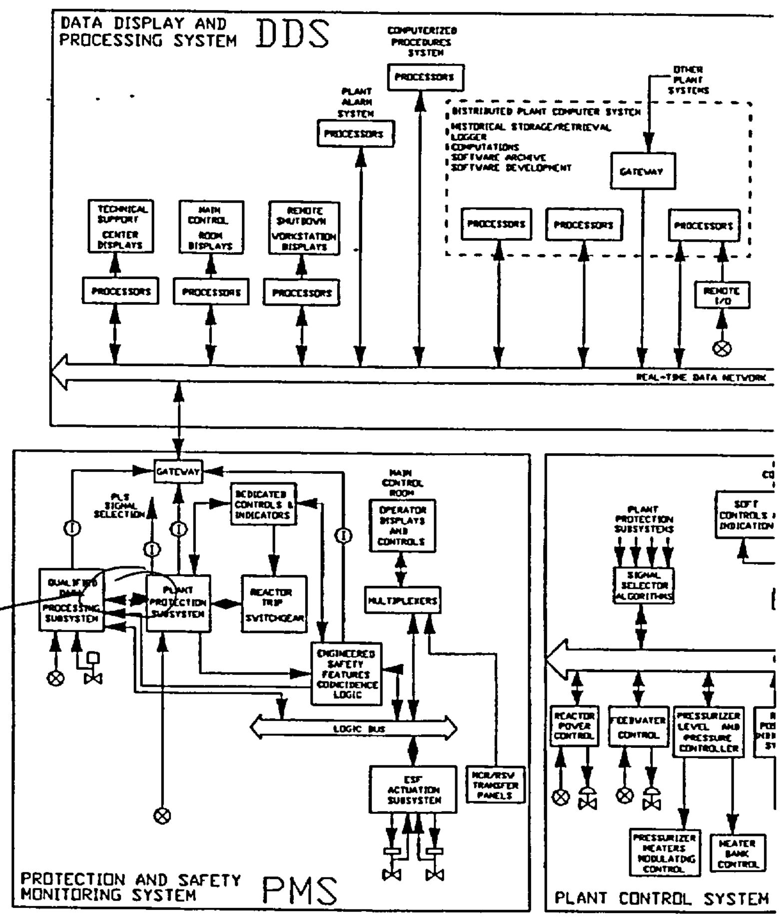
Design Control Document (DCD) Revision:

DCD Figures 7.1-1 and 7.1-2 will be revised to show two-way communication between the QDPS and the Plant Protection Subsystem. See attached mark-ups.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

REVISION to DCD Figure 7.1-1



add arrow to Plant Protection Subsystem

REVISION to DCD Figure 7.1-2

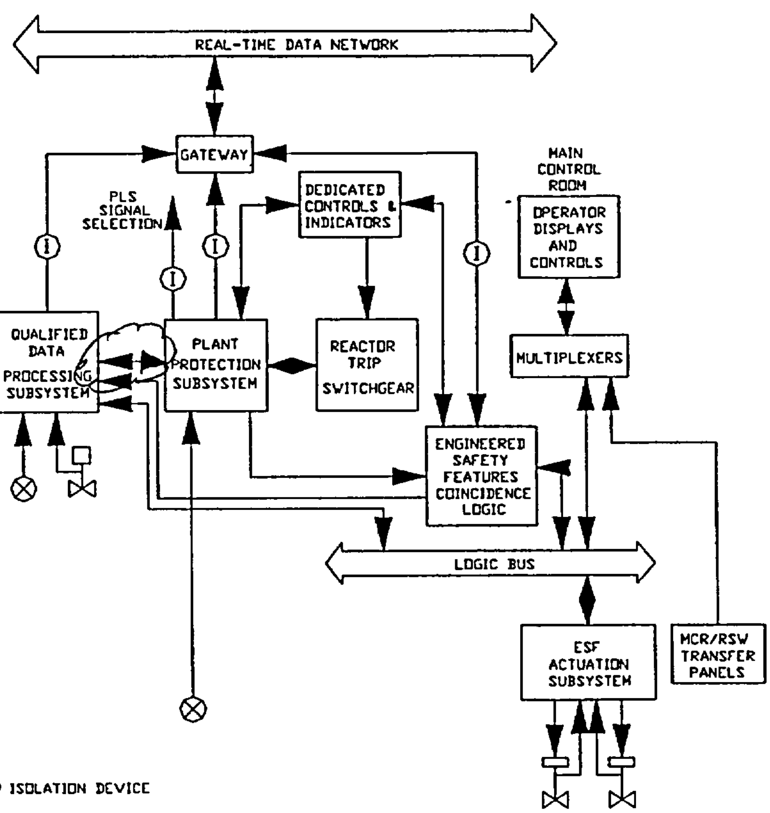


Figure 7.1-2

Protection and Safety Monitoring System

PRA Revision:

None

NRC Additional Comments:

The description stated in the Westinghouse response to RAI 420.007 should be included in the Design Control Document (DCD).

Westinghouse Additional Response:

DCD section 7.1.2.5 will be revised as shown below:

Design Control Document (DCD) Revision:

7.1.2.5 Qualified Data Processing Subsystems

The Qualified Data Processing Subsystem (QDPS), a subsystem of the PMS, provides safety-related display of selected parameters in the control room.

The QDPS subsystems are a redundant configuration consisting of sensors, QDPS hardware, and qualified displays.

The qualified data processing subsystems perform the following functions:

- Provide safety-related data processing and display
- Provide the operator with sufficient operational data to safely shut the plant down in the event of a failure of the other display systems
- Provide qualified and nonqualified data to the real-time data network for use by other systems in the plant
- Process data for main control room display, and to meet Regulatory Guide 1.97 requirements
- Provide data to the main control room, the remote shutdown workstation, the plant computer, other nonsafety-related devices, and nonqualified emergency response facilities in conformance with NUREG-0696

The QDPS hardware consists of safety-related modular data gathering units. The QDPS receives inputs from process sensors and safety-related digital systems. The QDPS consolidates the input data, performs conversions to process units, and formats the data for data link transmission.

Figure 7.1-8A illustrates the qualified data processing subsystem for the Eagle I&C architecture. Figure 7.1-8B illustrates the qualified data processing subsystem for the Common Q architecture.

Power is provided to the QDPS from the Class 1E dc and UPS system for 72 hours after a loss of all ac power (station blackout). After 72 hours, the ancillary diesel generators provide power for the QDPS. QDPS is a two-train subsystem (Divisions B and C). The PMS, including the QDPS, is diverse from the Diverse Actuation System (DAS). Sensors are not shared between PMS and DAS.

The RTS/ESFAS signals are processed by the Plant Protection Subsystem of PMS. Within PMS, some sensors are shared between the Plant Protection Subsystem and QDPS. Shared sensors are processed first by the QDPS because the QDPS will need this sensor for more than 24 hours following a station blackout. 24-hour batteries power the Plant Protection Subsystem; therefore, the Plant Protection Subsystem can not be used for QDPS functions.

The typical input parameter for RTS/ESFAS is four-way redundant with one sensor for each of the four divisions. If that parameter is also needed by QDPS, the B and C division sensors are processed first by QDPS then sent to the Plant Protection Subsystem. The A and D division sensors are not shared with QDPS and thus are processed directly by the Plant Protection Subsystem. If an RTS/ESFAS parameter is not needed by QDPS or if it is not needed after 24-hours, it is processed directly by the Plant Protection Subsystem in all four divisions.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.008 (Response Revision 1)

Question:

420.8 (DCD Figure 7.1-2)

Describe the "GATEWAY" design and its interface with the Protection and Safety Monitoring System.

Westinghouse Response:

The purpose of the Protection and Safety Monitoring Gateway is to interface the safety PMS to the non-safety real-time data network that supports the remainder of the instrumentation and control system. The Gateway has two subsystems. One is the safety subsystem that interfaces to the Plant Protection Subsystem, the Engineered Safety Features Coincidence Logic and the Qualified Data Processing Subsystem. The other is the non-safety subsystem that interfaces to the real-time data network. The two subsystems are connected by a fiber optic link that provides electrical isolation.

The primary flow of information between the two Gateway subsystems is from the safety subsystem to the non-safety subsystem. This information is a combination of plant process parameter values and equipment status information. The information that flows from the non-safety subsystem to the safety subsystem is limited to the following:

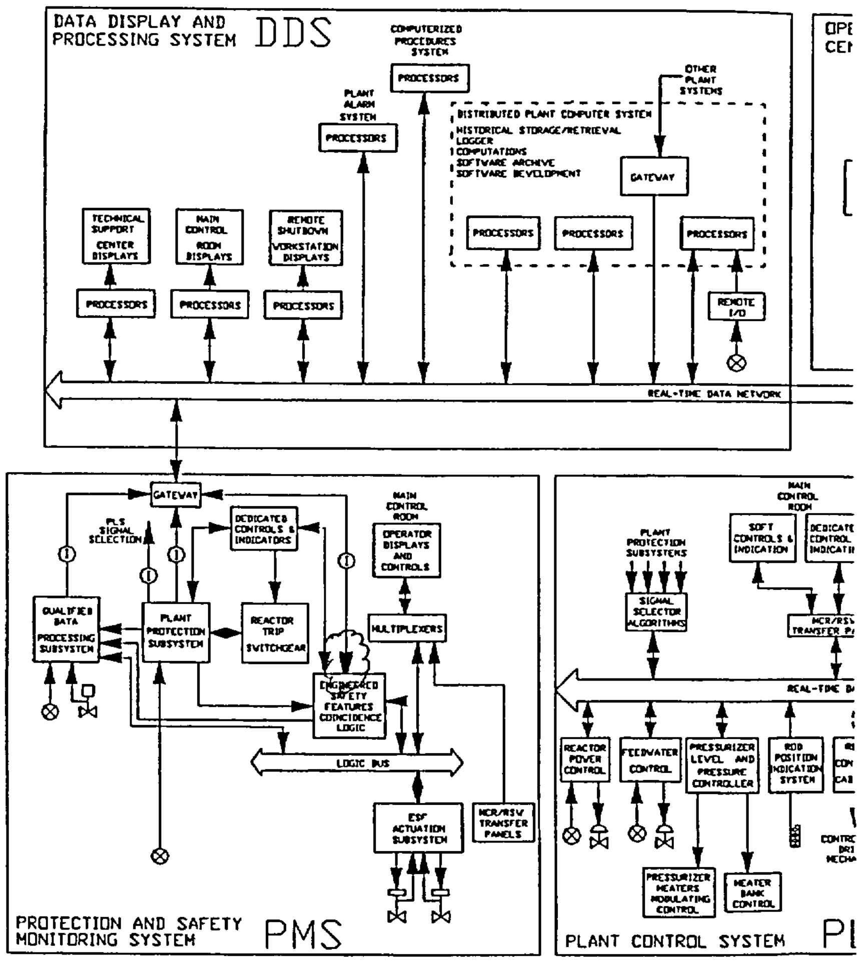
- The safety and non-safety subsystems exchange periodic interface signals that the communication controllers at each end of the link use to ensure that the link is functioning properly. These signals are used only by the communication controllers and are not propagated to the rest of the safety system. There is no application function in the safety system that uses this information.

- The main control room and the remote shutdown workstation operator consoles are non-safety. The soft control inputs to the PMS from these locations are provided from the non-safety subsystem to the safety subsystem of the Gateway.

Note that there is an error in DCD Figure 7.1-1. The gateway needs to communicate <u>to</u> the Engineered Safety Features Coincidence Logic (as well as from the ESF Coincidence Logic) to accomplish the second function listed above. DCD Figure 7.1-2 (Revision 1) shows this link correctly.

Design Control Document (DCD) Revision:

Revise DCD Figure 7.1-1 as shown.



PRA Revision:

None

NRC Additional Comments:

The description stated in the Westinghouse response to RAI 420.008 should be included in the DCD. In addition to the electrical isolation, Westinghouse should also address communication isolation.

Westinghouse Additional Response:

DCD section 7.1.2.8 will be revised as shown below:

Design Control Document (DCD) Revision:

7.1.2.8 Communication Functions

The communication functions provide information from the plant protection subsystem, the ESF coincidence logic, the ESF actuation subsystems, and the QDPS subsystems to external systems. This includes outputs to the plant control system and the data display and processing system. Isolation devices provide electrical isolation between the protection and safety monitoring system and the external systems.

The communication functions are accomplished via channelized gateways as shown in Figure 7.1-1.

The PMS Gateway interfaces the safety PMS to the non-safety real-time data network that supports the remainder of the instrumentation and control system. The Gateway has two subsystems. One is the safety subsystem that interfaces to the Plant Protection Subsystem, the Engineered Safety Features Coincidence Logic and the Qualified Data Processing Subsystem. The other is the non-safety subsystem that interfaces to the real-time data network. The two subsystems are connected by a fiber optic link that provides electrical isolation.

The primary flow of information between the two Gateway subsystems is from the safety subsystem to the non-safety subsystem. This information is a combination of plant process parameter values and equipment status information. The information that flows from the non-safety subsystem to the safety subsystem is limited to the following:

- **The safety and non-safety subsystems exchange periodic interface signals that the communication controllers at each end of the link use to ensure that the link is functioning properly. These signals are used only by the communication controllers and are not propagated to the rest of the safety system. There is no application function in the safety system that uses this information.**

- **The main control room and the remote shutdown workstation operator consoles are non-safety. The soft control inputs to the PMS from these locations are provided from the non-safety subsystem to the safety subsystem of the Gateway.**

The gateway provides both electrical and communication isolation between the non-safety systems and the PMS. Other than the isolation function, the gateway is not required for any PMS safety function. There is no potential signal from the non-safety system than will prevent the PMS from performing its safety functions.

Analog inputs required for both control and protection functions are processed independently with separate input circuitry. The input signal is classified as safety-related and is, therefore, isolated in the protection and safety monitoring system cabinet before being sent to the control system.

The plant protection and safety monitoring system also provides data to the plant control system pertaining to signals calculated in the subsystems, and to the data display and processing system.

Non-process signals are also provided to external systems. The non-process outputs inform the external systems of cabinet entry status, cabinet temperature, dc power supply voltages, and subsystem diagnostic status. Cabinet temperature sensing does not affect the safety-related function. The information is gathered for the sole purpose of analysis by external systems.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.013 (Response Revision 1)

Question:

420.13 (DCD 7.1.7, item 7)

DCD 7.1.7, item 7, WCAP-15775, "AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report," Section 4.11 states that the signal conditioning and data acquisition functions associated with these signals are performed by an independent subsystem in the PMS, not associated with the reactor trip or ESF actuation functions. Describe the system configuration with respect to this statement for both the AP600 system hardware and the Common Q system hardware.

Westinghouse Response:

The sentence cited from WCAP-15775 may be misleading and does not directly address NUREG/CR 6303 Guideline 13; therefore, WCAP-15775 will be modified as shown below.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

WCAP-15775 will be revised as shown:

4.11 Plant Monitoring – Guideline 13

Indications to support manual actions to maintain the plant within operating limits, trip the reactor, and actuate ESF functions are provided within the three layers of the instrumentation and control architecture. The DDS provides nonsafety operator displays and alarms. Plant data for the nonsafety displays and alarms is obtained from across the instrumentation and control architecture by means of the real-time data network. The QDPS within the PMS provides safety operator displays. In addition, the DAS provides nonsafety, diverse operator indications. No sensors are shared between the RTS/ESFAS and the DAS. Diverse and independent signal conditioning and data acquisition functions will be performed in the RTS/ESFAS and DAS such that a postulated software common mode failure in either platform will not degrade the signal conditioning and data acquisition functions in the other platform

Signals are transmitted from the PMS to the PLS and the DDS. The connections between the PMS and the PLS and DDS contain isolation devices to prevent failures in the PLS or DDS from affecting operation of the PMS. Once signals leave the PMS through the isolation devices, they are no longer classed as safety, and are not used to provide any safety functions.

The signals from PMS to PLS and DDS meet the independence requirements of GDC-24, IEEE-603, IEEE-379, and IEEE-384.

No credible failure of the PLS or DDS will prevent the safety system from performing its safety function. Connections and software that are used for plant monitoring and for surveillance of the reactor trip and ESF actuation system do not significantly reduce the reliability of or increase the complexity of these systems.

The automatic functions of the PMS are designed to protect the AP1000 from potential operator-induced transients which may result from failures in the DDS or PLS.

NRC Additional Comments:

Please clarify was is meant by the phrase "do no significantly reduce" in the statement in the response that states that "[c]onnections and software that are used for plant monitoring and for surveillance of the reactor trip and ESF actuation system do not significantly reduce the reliability of or increase the complexity of these systems."

Westinghouse Additional Response:

WCAP-15775 will be revised as shown below:

WCAP Revision:

WCAP-15775 will be revised as shown:

4.11 Plant Monitoring – Guideline 13

Indications to support manual actions to maintain the plant within operating limits, trip the reactor, and actuate ESF functions are provided within the three layers of the instrumentation and control architecture. The DDS provides nonsafety operator displays and alarms. Plant data for the nonsafety displays and alarms is obtained from across the instrumentation and control architecture by means of the real-time data network. The QDPS within the PMS provides safety operator displays. In addition, the DAS provides nonsafety, diverse operator indications. No sensors are shared between the RTS/ESFAS and the DAS. Diverse and independent signal conditioning and data acquisition functions will be performed in the RTS/ESFAS and DAS such that a postulated software common mode failure in either platform will not degrade the signal conditioning and data acquisition functions in the other platform.

Signals are transmitted from the PMS to the PLS and the DDS. The connections between the PMS and the PLS and DDS contain isolation devices to prevent failures in the PLS or DDS from affecting operation of the PMS. Once signals leave the PMS through the isolation devices, they are no longer classed as safety, and are not used to provide any safety functions.

The signals from PMS to PLS and DDS meet the independence requirements of GDC-24, IEEE-603, IEEE-379, and IEEE-384.

No credible failure of the PLS or DDS will prevent the safety system from performing its safety function. **The Gateway provides the** connections that are used for plant monitoring and for surveillance of the reactor trip and ESF actuation **subsystems. The DDS provides the software and hardware that are used for displaying plant parameters and monitoring system performance.**

The automatic functions of the PMS are designed to protect the AP1000 from potential operator-induced transients which may result from failures in the DDS or PLS.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.045 (Response Revision 1)

Question:

420.45 (DCD Tier 1, Section 2.5.2)

Describe the method of safety analysis that will be performed on the PMS and its components to ensure the PMS will perform as specified and that failure have been identified.

Westinghouse Response:

Several activities will be performed to validate that the protection and safety monitoring system (PMS) will meet the stated functional requirements. These include the following:

a. Subjecting the PMS to environmental and seismic testing or referencing the results of applicable tests that have already been conducted, on the product line (ITAAC 2.5.2, items 2 and 4),
b. Subjecting the PMS to Electromagnetic Interference (EMI) testing or referencing the results of applicable tests that have already been conducted on the product line (ITAAC 2.5.2, item 3),
c. Performing a system level Failure Mode and Effects Analysis (FMEA) to demonstrate the PMS architecture is not susceptible to postulated single random failures (COL item, see response to RAI 420.028),
d. Use of a structured software life cycle process, per IEEE Std. 7-4.3.2, to minimize the possibility of software common mode failure errors (ITAAC 2.5.2, item 11), and
e. Subjecting the PMS to a Factory Acceptance Test (FAT) and Site Acceptance Test (SAT) to validate the system meets the stated functional requirements (ITAAC 2.5.2, items 5, 6, 8, and 9),
f. Performing preoperational tests as described in DCD subsection 14.2.9.1.12,
g. Performing startup tests as described in DCD subsections 14.2.10.1.8 and 14.2.10.1.10.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

The staff does not consider RAI response a,b,e,f, and g to be methods of safety analysis and, while important steps in the engineering life-cycle, are not considered methods of safety analysis. Items c and, to some degree, d meet the intent of the RAI with follow-up questions. A system level failure modes and effects analysis (FMEA) may be insufficient in detail to determine failure modes of such components as the gateway—and its components, the communication interfaces used between channels and isolation devices. What level of detail is considered sufficient for the proposed system level FMEA? Please provide a rationale for the level of detail

Westinghouse agreed to revise the RAI response and the DCD to include a Software Hazards Analysis as part of the COL action item to perform an FMEA.

Westinghouse Additional Response:

The FMEA will include a Software Hazards Analysis. DCD section 7.2.3 will be revised as shown below:

Design Control Document (DCD) Revision:

7.2.3 Combined License Information

Combined License applicants referencing the A1000 certified design will provide an FMEA for the protection and safety monitoring system. **The FMEA will include a Software Hazards Analysis.**

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.043 (Response Revision 1)

Question:

Section 5.4.2.1 states that Chapter 15 discusses the accident analysis of a steam generator tube rupture, which is based on a rupture of one tube.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 12, 1993, the NRC staff states its position that an applicant for a passive pressurized-water reactor (PWR) design certification should assess features to mitigate the amount of containment bypass leakage that could result from the rupture of multiple steam generator (SG) tubes. This position arises from a concern that an multiple-tube rupture event creates the likelihood of a SG safety valve (SGSV) lifting and then failing to close, resulting in an unisolable release to the environment bypassing the containment.

A. Discuss the AP1000 design features that mitigate or prevent SGSV challenges during an event of rupture of multiple steam generator tubes.

B. Provide an analysis of multiple-tube rupture events to address the concern of containment bypass leakage resulting from a potential failure of the SGSV to reclose.

Westinghouse Response: (Revision 1)

A. The objectives of the AP1000 plant response to multiple steam generator tube rupture are the same as for the AP600, including:

- To automatically terminate the loss of reactor coolant (RC) without overfilling the steam generator (SG) or opening the secondary safety valves. Operation of the automatic depressurization system (ADS) should not occur during these accidents.
- In the unlikely event of a secondary safety valve failing open, to provide defense-in-depth core cooling through the use of the ADS and passive safety injection.

The AP1000 passive systems provide a unique response to the SGTR initiating event with respect to conventional plants by automatically terminating the loss of reactor coolant without **actuating the ADS valves** or overfilling the SG. The passive residual heat removal (PRHR) heat exchanger acts to reduce the RCS pressure below the pressure of the secondary system and isolate the break flow to the faulted SG. The heat is removed from the RCS through the PRHR instead of the intact SG PORV to stop the leak to the faulted SG. The core makeup tanks (CMTs) provide heat removal and coolant inventory makeup for shrinkage in the RCS. During a SGTR, the CMTs inject water in the recirculation mode, exchanging cold borated water for hot RCS water. The CMTs do not drain during recirculation injection, and therefore, automatic depressurization system (ADS) is not actuated. Low CMT level is the initiation setpoint for ADS.

The AP1000 also provides additional defense-in-depth to mitigate multiple SGTRs. The active, nonsafety related systems can be used to mitigate the multiple SGTR as in a conventional plant, however, in the AP1000 this is not the safety case presented in the SSAR. The intact SG PORV is used to control the RCS pressure and isolate the break. The Chemical and Volume Control System (CVS) auxiliary spray is used to reduce the RCS pressure to allow the pumped residual heat removal system to provide borated makeup flow to the system until the break is isolated. In case of failure of both the active nonsafety related mitigation and also the passive safety related PRHR HX mitigation, the AP1000 provides another defense-in-depth method of mitigation. This method uses the ADS and passive safety injection.

If the secondary system safety valve is arbitrarily assumed to fail open, the PRHR HX will not be able to terminate the loss of RC. The loss of primary system coolant through the SG tube and the stuck open valve eventually causes the CMTs to drain to the ADS actuation setpoint. Actuation of ADS depressurizes the RCS in a controlled, staged manner. The safety related passive injection systems, CMTs, accumulators and IRWST gravity injection provide inventory makeup and boration throughout the depressurization. The core remains covered and cooled through out the sequence, and the plant achieves a safe, stable configuration without a release of fission products from the fuel matrix. Preventing the release of fission products from core mitigates the beyond-design-basis containment bypass.

B. Multiple SG tube ruptures are considered to be beyond design basis accidents. Analyses of two 5-tube multiple-SGTR cases for the AP1000 have been performed with the MAAP4.04 code using best estimate decay heat and best estimate assumptions, except as noted below. The cases are based on the cases presented for the AP600 in reference 1.

B.1 Case 1 – Multiple SGTR with Passive System Response

Case 1 is a passive system mitigation case with PRHR heat operation. The accident is initiated by the simultaneous double-ended failure of five cold side tubes at the top of the tubesheet. Startup feedwater (SFW) and the CVS are conservatively assumed to function because they tend to make the accident worse. The SFW controls operate normally and throttle the startup feedwater based on the nominal SG operating level. The CVS provides RCS makeup until it is isolated on a hi-2 SG narrow range level. The secondary system PORV is assumed to not open.

The MAAP4 results are presented in Figure 440.43-1 through 440.43-9. The results show that the faulted SG does not overfill and the safety valves do not open. Therefore, bypass does not occur.

B.2 Case STK – Multiple SGTR with Failed Open SG Safety Valve

Case STK is a passive system mitigation case with minimum PRHR heat removal. The accident is initiated by the simultaneous double-ended failure of five cold side tubes at the top of the tubesheet. SFW and the CVS are conservatively assumed to function because they tend to make the accident worse. The SFW control is assumed to malfunction such that the SFWS flow continues when the SG level increases above the normal level until it is isolated by a hi-2 SG narrow range level. . The CVS provides RCS makeup until it is isolated on a hi-2 SG narrow range level. The secondary system PORV is assumed to not open. The combination of the low PRHR heat removal and the high SG level control causes the faulted SG pressure to exceed the safety valve setpoint. When the valve opens, it is assumed to fail open although the SG is not predicted to overfill.

The MAAP4 results for case STK are presented in Figure 440.43-10 through 440.43-18. The loss of coolant from the RCS eventually drains the CMTs to the ADS actuation setpoint. The RCS depressurizes and gravity injection begins. The core remains covered and cooled, thus no significant fission product release occurs. Boron dilution due to secondary system water ingress in the RCS during depressurization is not a problem because the PXS injection tanks (CMTs, Accumulators and IRWST) provide boron to the RCS and because boron collects in the secondary system prior to ADS (reference 1). Therefore, the boron concentration of the water coming back into the RCS is expected to have approximately the same boron concentration as the water in the RCS.

References

1. WCAP-14991, AP600 Multiple Steam Generator Tube Rupture Analysis Report.

NRC Additional Comments:

Response to item (A) states that the AP1000 passive systems provide a unique response to the SGTR initiating event with respect to conventional plants by automatically terminating the loss of reactor coolant without depressurizing the RCS [reactor coolant system] or overfilling the SG.

Since the RCS is actually depressurized through the PRHR HX operation in response to an SGTR in AP1000, clarify the statement ".... without depressurizing the RCS..."

Westinghouse Revised Response:

The Revised Response provided above has been clarified.

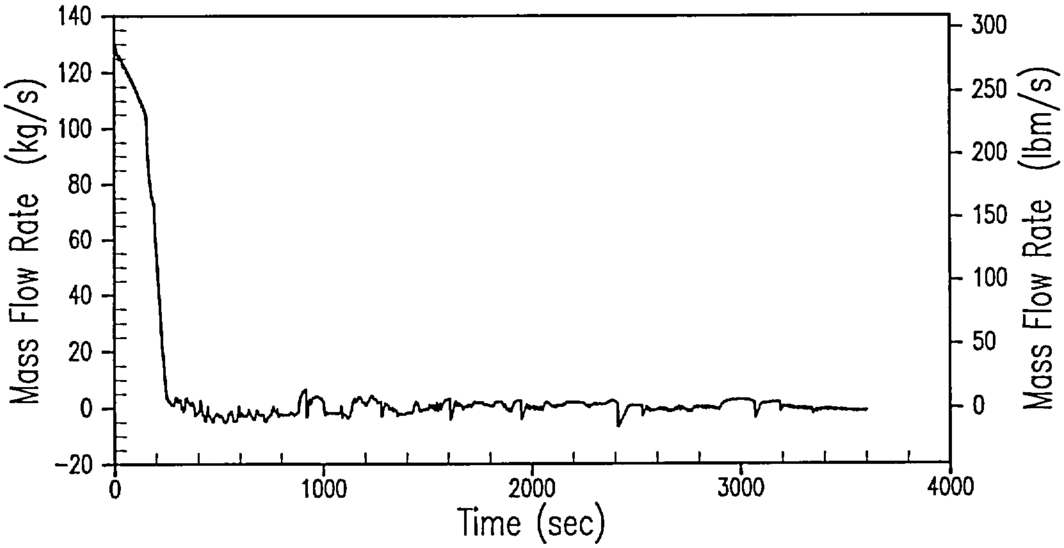


Figure 440.43-1 AP1000 Multiple (5 Tubes) SGTR with PORV Failure Tube Rupture Break Flow

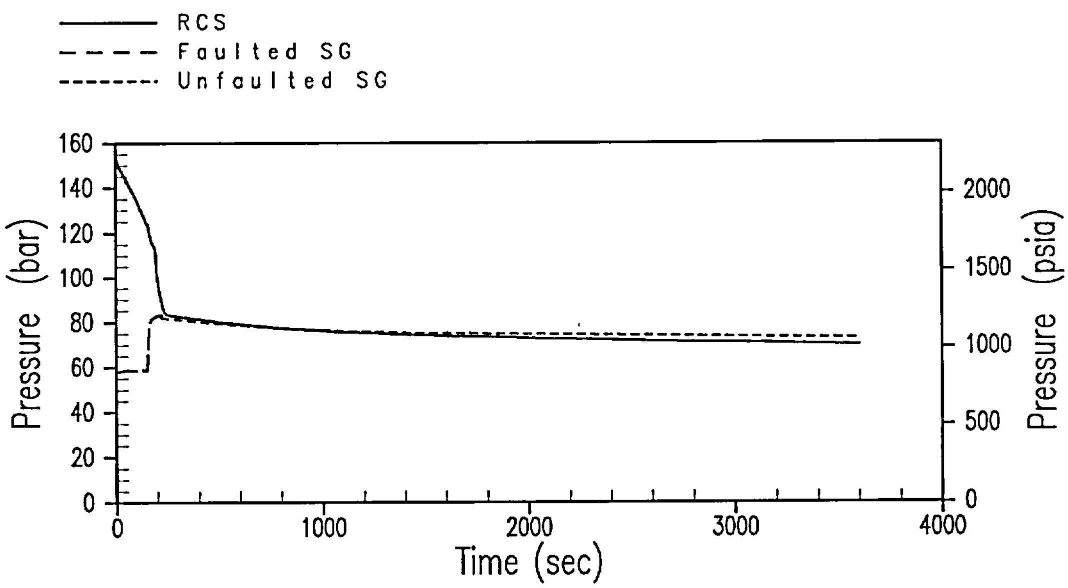


Figure 440.43-2 AP1000 Multiple (5 Tubes) SGTR with PORV Failure RCS and Secondary System Pressures

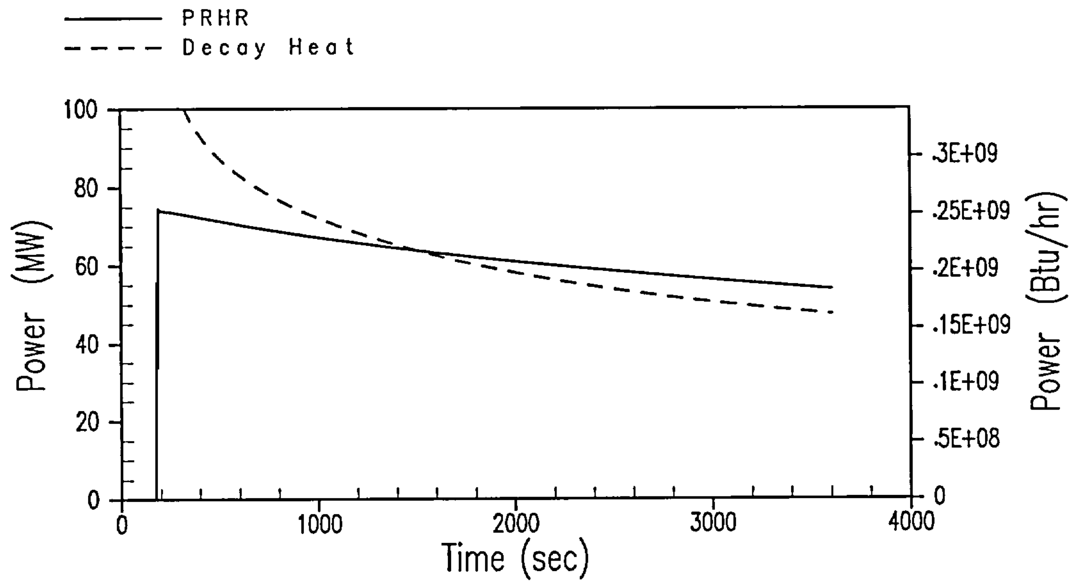


Figure 440.43-3 AP1000 Multiple (5 Tubes) SGTR with PORV Failure PRHR Heat Removal

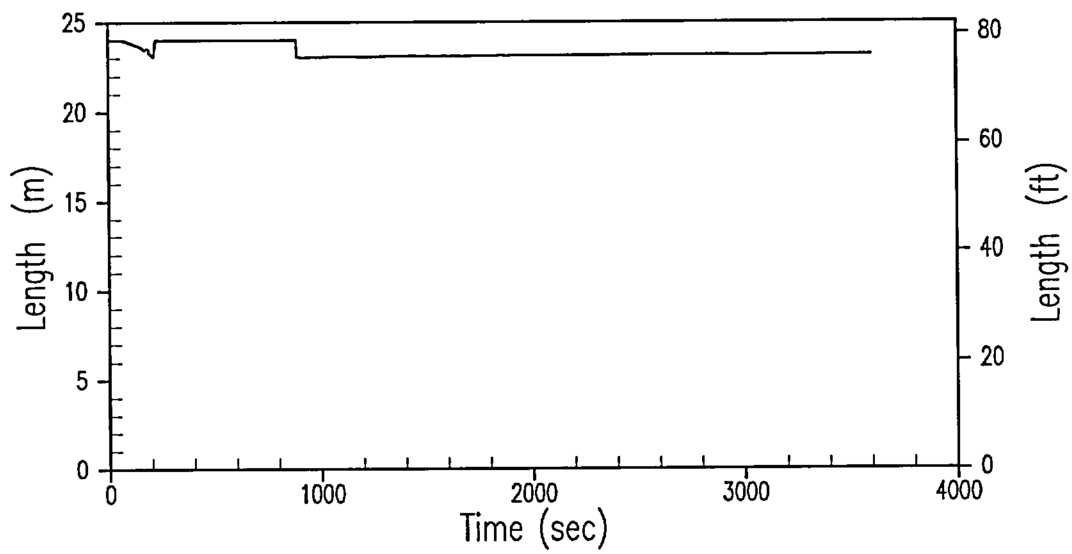


Figure 440.43-4 AP1000 Multiple (5 Tubes) SGTR with PORV Failure RCS Water Level

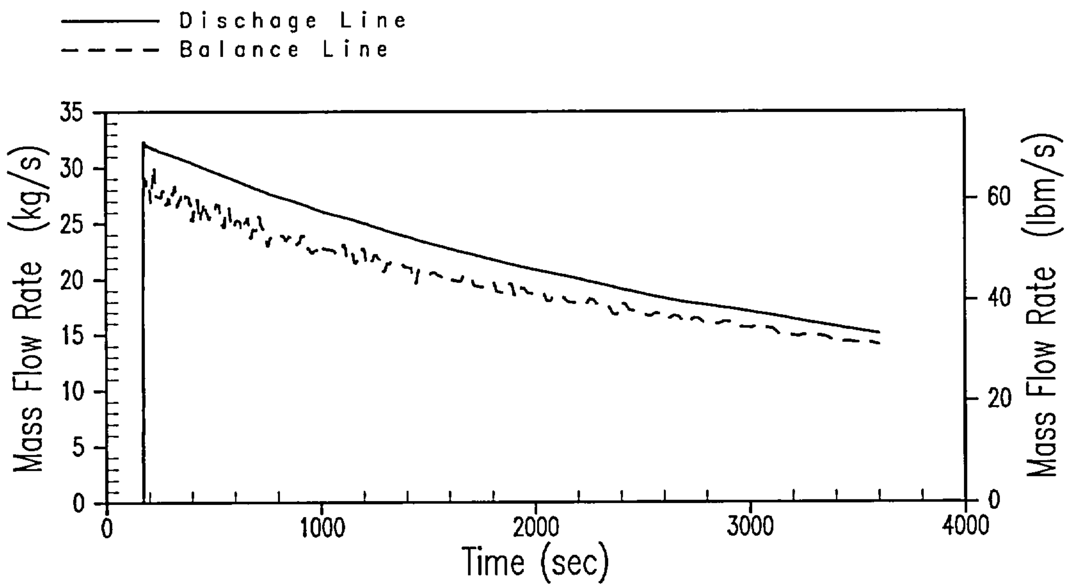


Figure 440.43-5 AP1000 Multiple (5 Tubes) SGTR with PORV Failure CMT Water Mass Flow Rates

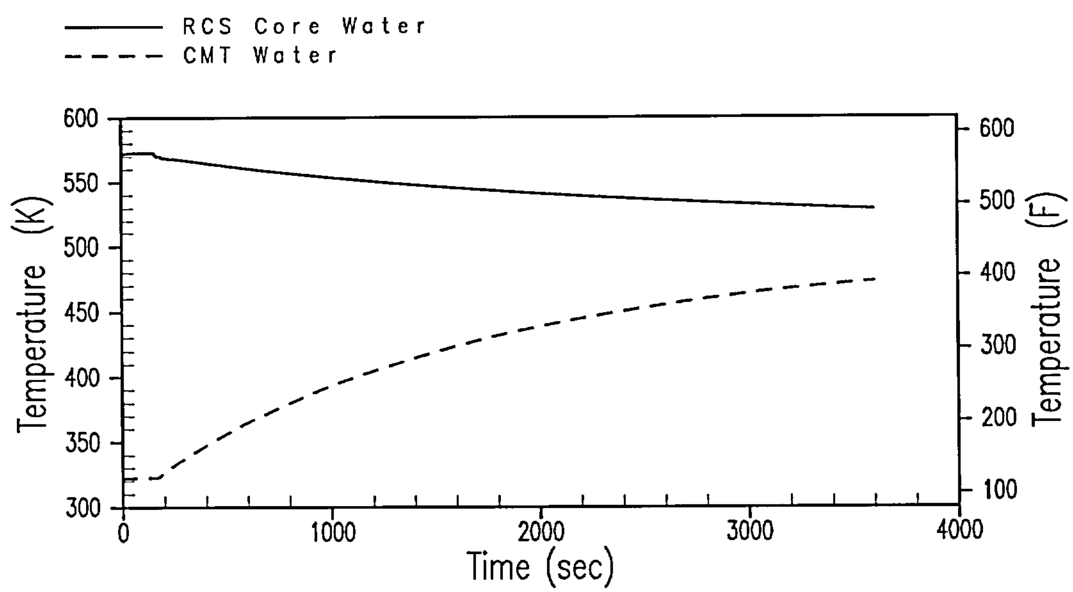


Figure 440.43-6 AP1000 Multiple (5 Tubes) SGTR with PORV Failure RCS and CMT Water Temperatures

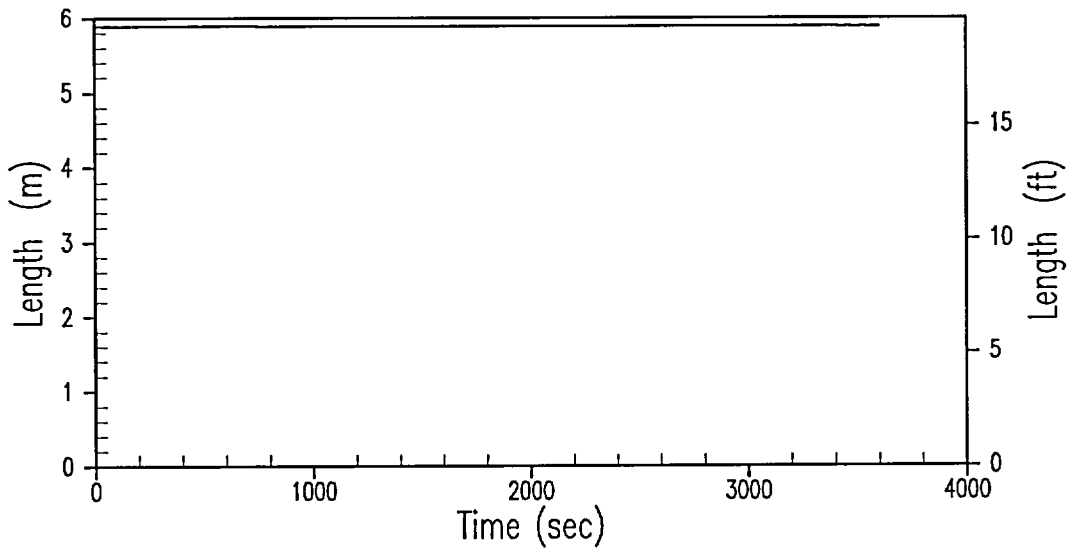


Figure 440.43-7 AP1000 Multiple (5 Tubes) SGTR with PORV Failure CMT Water Level

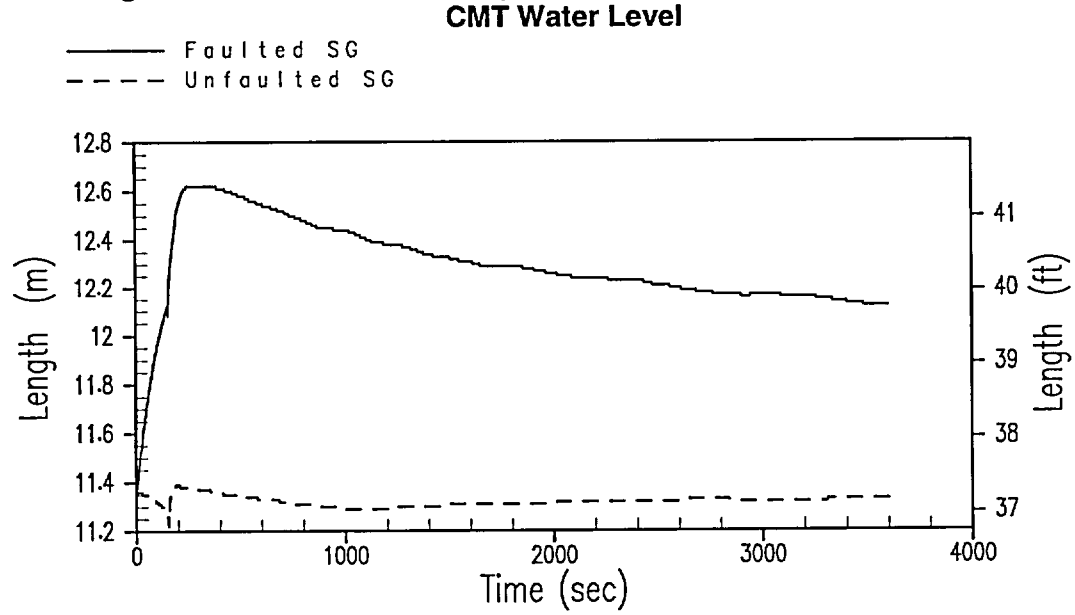


Figure 440.43-8 AP1000 Multiple (5 Tubes) SGTR with PORV Failure Steam Generator Downcomer Water Level

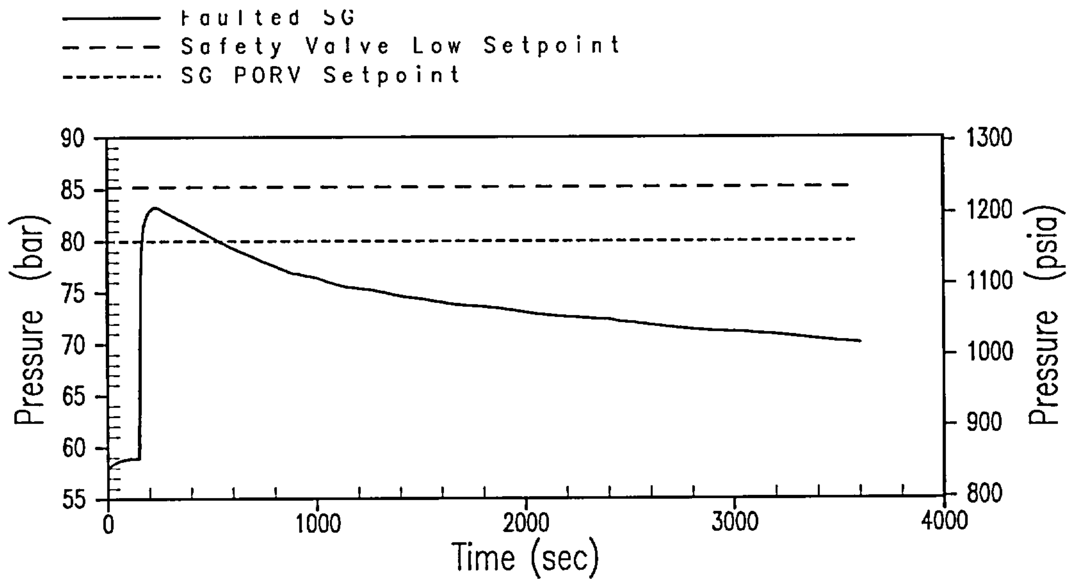


Figure 440.43-9 AP1000 Multiple (5 Tubes) SGTR with PORV Failure Faulted Steam Generator Pressure

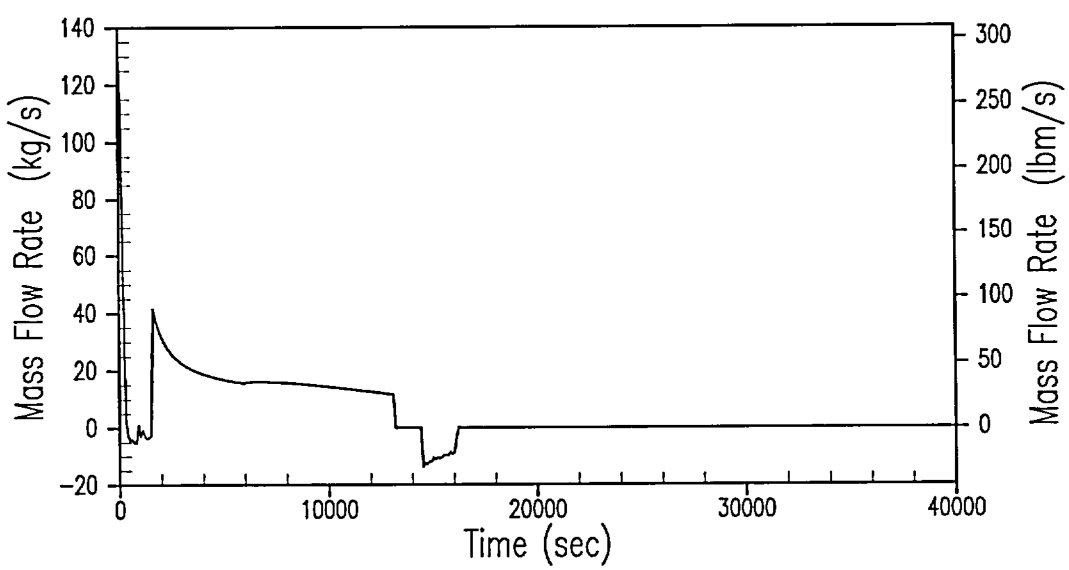


Figure 440.43-10 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve Tube Rupture Break Flow

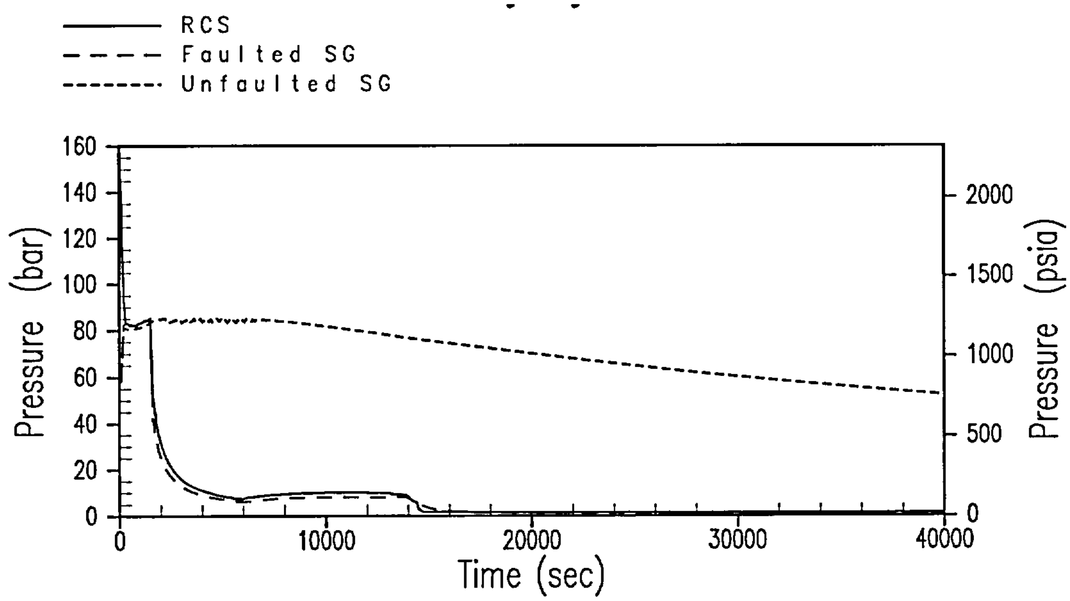


Figure 440.43-11 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve RCS and Secondary System Pressures

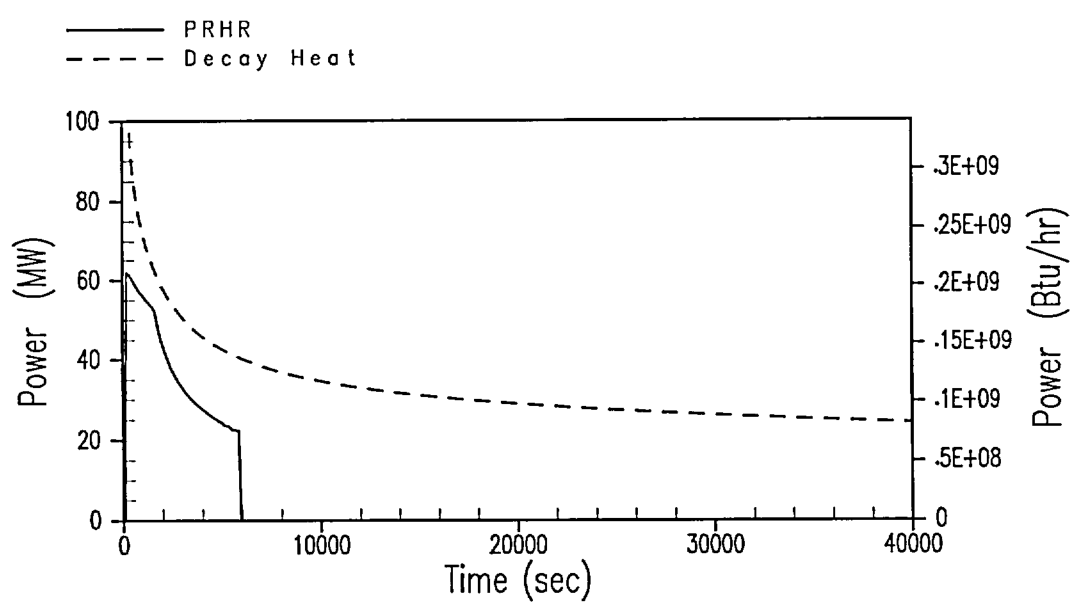


Figure 440.43-12 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve PRHR Heat Removal

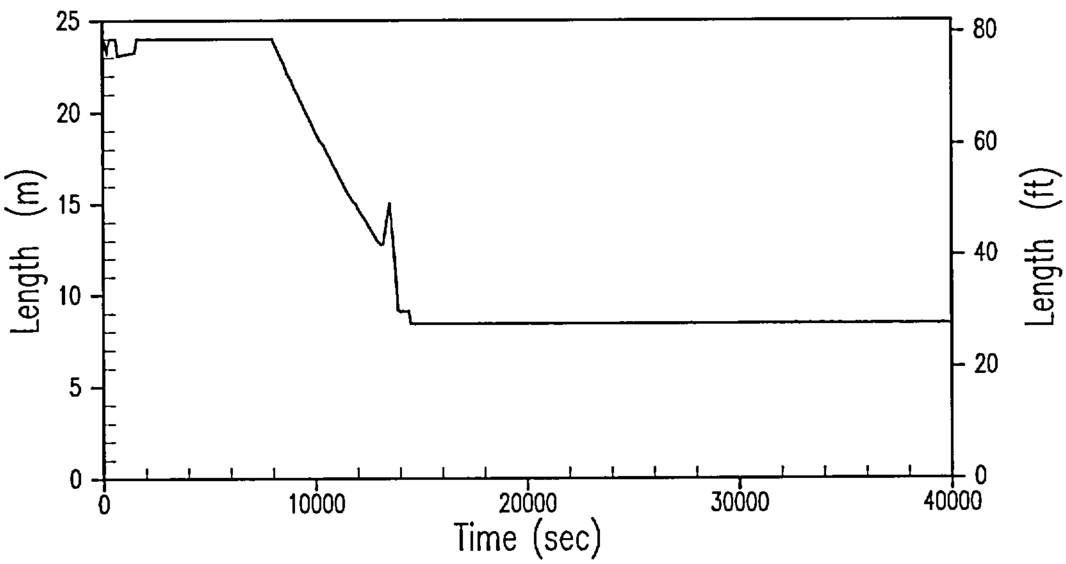


Figure 440.43-13 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve RCS Water Level

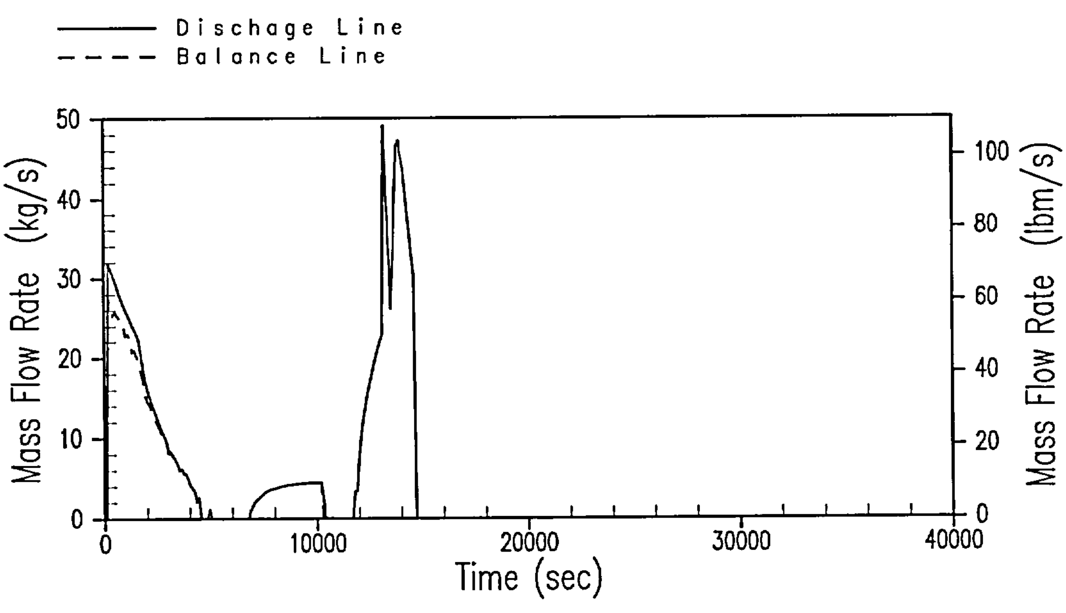


Figure 440.43-14 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve CMT Water Mass Flow Rates

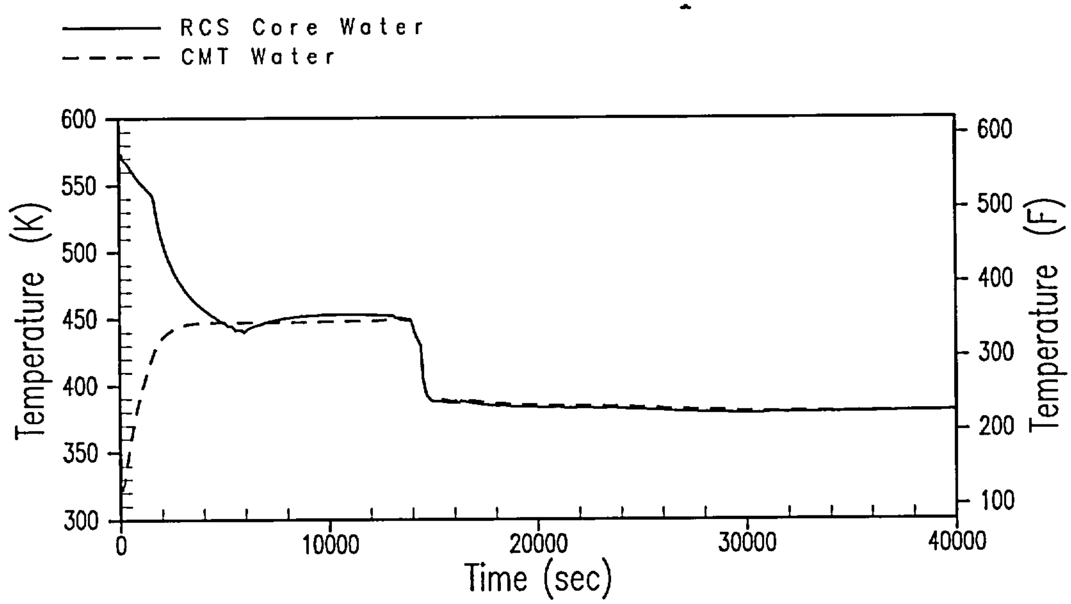


Figure 440.43-15 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve RCS and CMT Water Temperatures

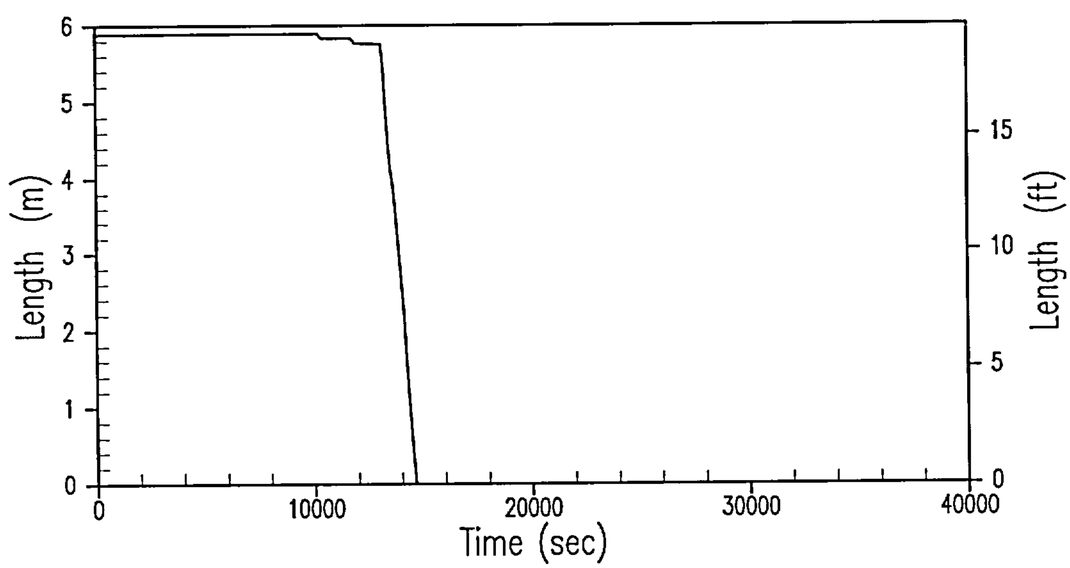


Figure 440.43-16 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve CMT Water Level

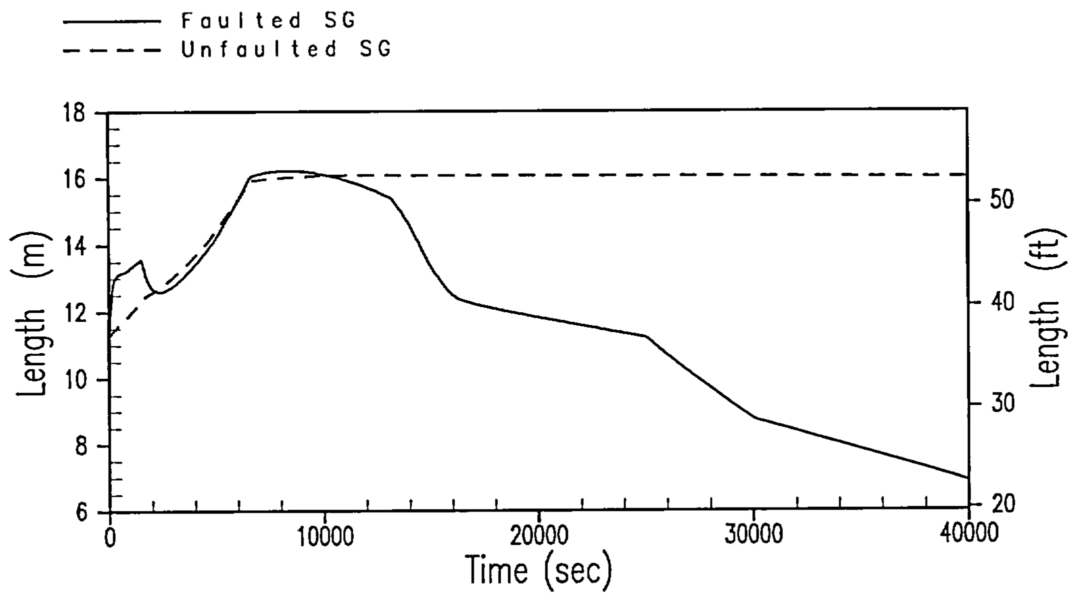


Figure 440.43-17 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve Steam Generator Downcomer Water Level

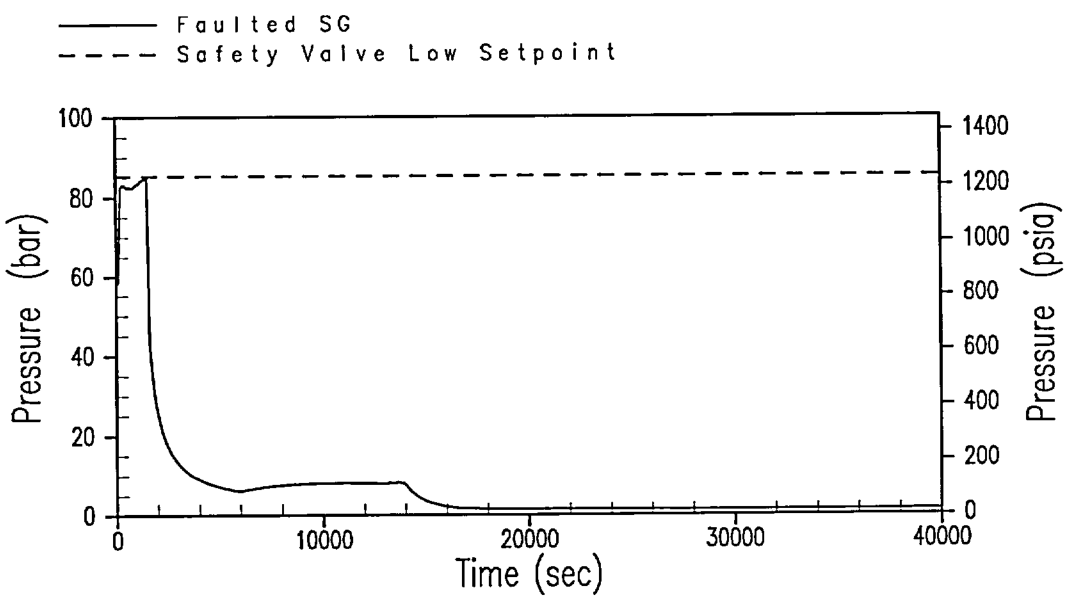


Figure 440.43-18 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve Faulted Steam Generator Pressure

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.050 (Response Revision 1)

Question:

In Tier 2 Information, Section 6.3 describes the passive core cooling system. Each of the core makeup tank (CMT) and accumulator outlet injection lines contains a flow-tuning orifice that provides a mechanism for the field adjustment of the injection line resistance to establish the required flow rates assumed in the design. In the AP600 design, flow tuning orifices are also included in the in-containment refueling water storage tank (IRWST) injection lines.

Provide the reason why the AP1000 IRWST injection lines do not have flow-tuning orifices.

Westinghouse Response:

The flow tuning orifices in the AP600 IRWST injection lines had a limited function. The IRWST line B had a portion of 10" pipe from the IRWST to the RNS pump suction connection. The rest of the line is 8". All of line A is 8". As a result, line B would of had a lower line resistance than line A. The only function of the orifices were to limit the minimum resistance of line B to be no less than the minimum resistance of line A. This tuning minimized the potential spill in case of a DVI break in line B.

For AP1000 both lines have the same diameters; both have a 10" section connected to the IRWST that changes to 8" just before the connection with the recirculation lines. As a result the resistances of the two lines are nearly the same. In addition, elimination of the orifices eliminates large flanges, which helps to accommodate the larger pipe size in the AP1000 (from a physical space view). The minimum and maximum line resistances have been calculated for these two lines. This calculation accounts for not having flow tuning orifices. Those resistances have been used in the safety analysis performed for the DCD and they have been used in the ITAAC acceptance criteria.

NRC Additional Comments:

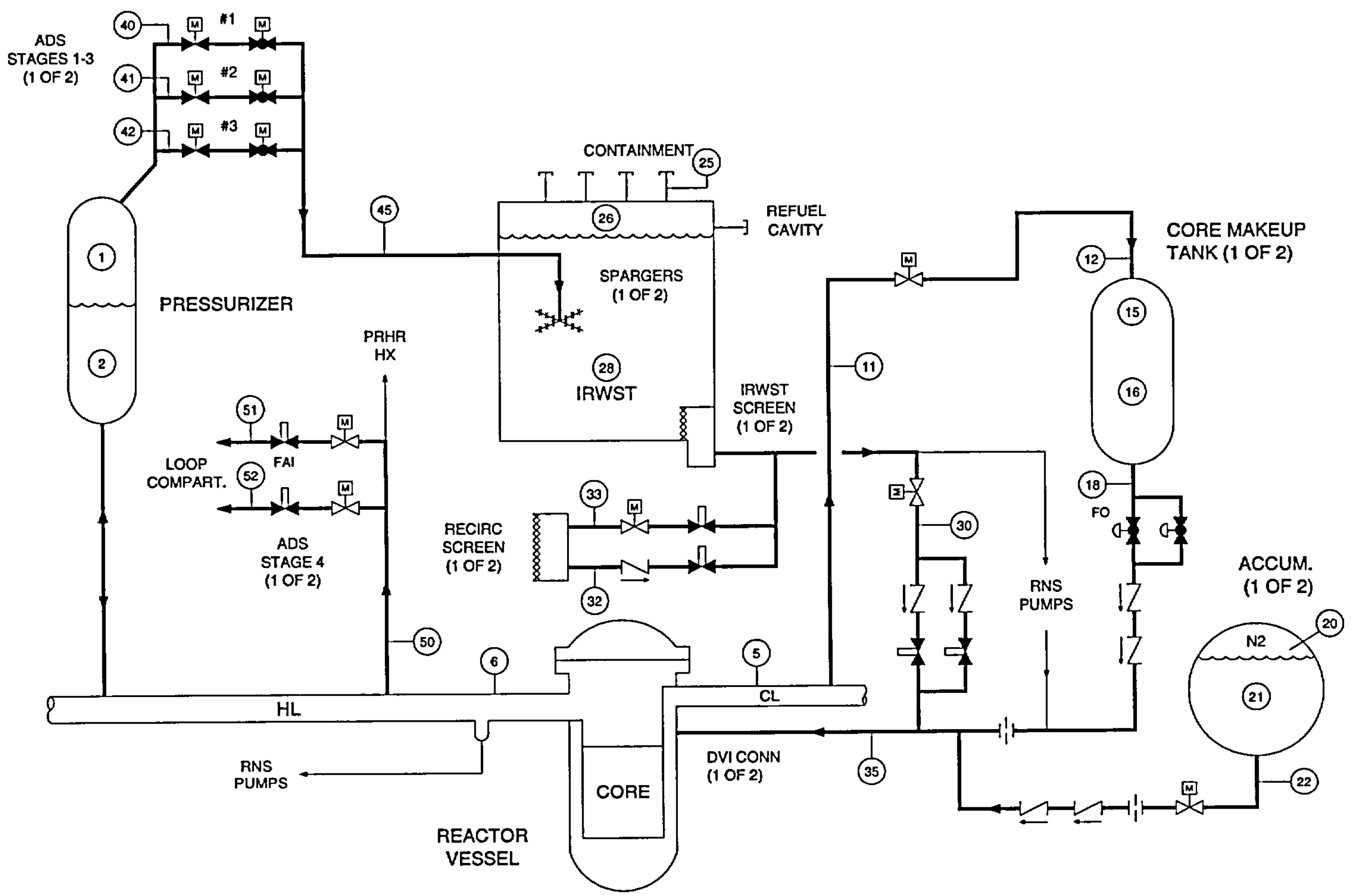
The response to RAI 440.050 explained the reason for not having an flow-adjusting orifice in the IRWST injection line. However, design control document (DCD) Figure 6.3-3 still shows an orifice in the IRWST injection line (Figure 6.3-2 does not show a flow orifice).

Westinghouse Revised Response:

The DCD Figure 6.3-3 will be revised to delete the flow tuning orifice in the IRWST injection line.

Design Control Document (DCD) Revision:

DCD Figure 6.3-3 will be replaced with the following figure.



PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.128 (Response Revision 1)

Question:

NUREG-0933, A Prioritization of Generic Safety Issues," Task Action Plan Item USI A-17 addresses the concerns of adverse systems interactions (ASI) among various structures, systems, and components (SSC) in a plant, and identifies the need to investigate the possibility that unrecognized subtle dependencies among the SSCs have remained hidden and could lead to safety significant events. The staff concluded that occurrence of an actual ASI or the existence of a potential ASI, as well as the potential overall safety impact, are very much a function of an individual plant's design and operational features. Therefore, for new plant designs with new or different configured passive and active systems, such as AP600 and AP1000 designs, the staff believes the designer should perform a systematic search for ASIs, and propose resolutions for any that are discovered. For the AP600 design, Westinghouse submitted topical report WCAP-14477, Revision 1, "The AP600 Adverse System Interaction Evaluation Report," to identify possible adverse interactions among safety-related systems and between safety-related and non-safety-related systems, and to evaluate the potential consequences of such interactions.

Provide a systematic evaluation of the ASI for the AP1000 design, similar to WCAP-14477, Revision 1, or provide detailed justifications, considering the differences between AP600 and AP1000, on why the ASI evaluation performed for the AP600 design and conclusion are applicable to AP1000.

Westinghouse Response:

Systematic evaluation of the ASI for the AP1000 design is provided in WCAP-15992, Rev. 0, "AP1000 Adverse System Interactions Evaluation Report".

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

1. Section 2.2.1, Reactor Coolant Pump Interaction (P. 2-10, 2nd paragraph):

   It states that " the AP1000 has addressed this (RCP - CMT) Interaction by providing a safety-related automatic trip of the RCPs on CMT actuation. In addition, as stated earlier, restart of the RCPs is not performed unless the CMT termination criteria can be met."

   Please clarify where this was stated. Clarify what are the CMT termination criteria? What are the indications for the plant to have been brought to stable safe condition for CMT termination?

2. Section 2.3.1.5, CMT/Accumulator - ADS Interaction (P. 2-37):

   It states that " in sensitivity studies performed to verify PRA success criteria, a phenomena was evaluated for cases of multiple failures in the ADS, where additional injection resulted in a higher (although acceptable) PCT. Ultimately, these multiple ADS failures were not credited as success in the PRA.... additional CMTs or accumulators do not adversely impact the ability of the ADS to depressurize the RCS...."

   Does that mean that the CMT and accumulator injection have no adverse interaction effect on the 3 out of 4 ADS-4 success criteria, but have potential adverse effect on the cases with less than 3 ADS-4 success? What are the effects on ADS 1,2, and 3 success criteria?

3. Section 2.3.4.1 Containment recirculation - PRHR HX interaction (P. 2-46):

   Section 2.3.4.1 indicates that there is no significant interaction between containment recirculation and PRHR.

   Since the containment recirculation paths contain a normally open MOV and a squib valve. Could there be an adverse effect on the PRHR HX of a spurious opening of a squib valve, which could result in the IRWST draining to the containment?

4. P. 2-50, Section 2.3.5.5 PRHR HX - RCS Interaction

   Would there be an adverse effect on the RCS piping and components of the PRHR HX operation that results in thermal stratification in the RCS?

5. P. 3-16, Item 2.3.6.4 ADS Actuation during Power Operation:

   It states that a spurious actuation of the ADS during power operation would lead to a LOCA event. This event is explicitly modeled in the AP1000 PRA.

   Do the PRA analyses cover both the spurious actuation of ADS-4 valves, spurious actuation of stages 1, 2 and 3 valves, or both?

6. Typos:

   P.2-1, 4th paragraph, P. 2-29, 5th para., and P. 3-15, 3rd para.:
   "AP600" should be "AP1000"?
   P. 3-3, 1st para.: "In Section 4 ...." should be "In section 3.4...."?
   P. 3-3, 3rd para.: "...in subsection 3.3.1..." should be "...in subsection 3.1 ..."?
   Table 2-2, item 15.4.1 is repeated.

7. WCAP-15922 section 3, "Evaluation of Potential Human Commission Errors," does not address the possibility of a plant malfunction (such as failed indication) inhibiting an operator's ability to respond. (The Data Display and Processing system (DDS) is a non-safety related computer-based system that can be failed.) What is the backup provision to support the analyses described in the WCAP-15992?

Westinghouse Revised Response:

WCAP-15992, Rev. 0, "AP1000 Adverse System Interactions Evaluation Report" will be revised to address the above comments as follows.

1. In accordance with the Emergency Response Guidelines (ERGs), CMT termination criteria include RCS subcooling and pressurizer water level. This is discussed in report section 2.2.1 (pg. 2-9 Para 4). No WCAP revision required.

2. There is no adverse interaction for the AP1000 ADS success sequences and CMT/accumulator injection flow. AP1000 success criteria are 3 of 4 ADS valves and both accumulators. Refer to RAI 720.026, which discusses MAAP4 analyses performed for an AP1000 2-inch HL break with 1 CMT and 2 CMTs. The additional water from the second CMT slightly delays the system transient, and results in less core uncovery.

   The writeup in the WCAP is referring to success criteria evaluations performed for AP600 multiple failure cases that ultimately were not considered success. The WCAP will be revised to delete the 4th paragraph in section 2.3.1.5, which is not applicable to the AP1000 design.

3. The following will be added to section 2.3.4.1:

   "Spurious opening of the long term recirculation flowpath squib valve, which could result in IRWST draining to containment and loss of PRHR cooling, is not a credible interaction. The component level controls provided for the long term recirculation flowpath squib valve are arranged in two separate locations within one I&C division. These controls perform separate "arm" and "fire" functions. They have to be actuated correctly in sequence in order for the squib valve to be actuated. As a result, there is no single active failure in these component controls that can cause the valve to open. In addition, actuation signals can come from automatic system level signals or from manual signals. The automatic signals are generated using 2 out of 4 logic. The manual signals are generated by soft controls or by dedicated switches. Soft controls are not considered subject to single failures that would generate spurious signals. The dedicated switches require two separate switches to be actuated at the same time."

4. Thermal stratification is considered appropriately in the design of the AP1000 piping systems including the loop piping and PRHR piping. Westinghouse responses to RAIs 210.017, 048, 049 and 050 as well as AP1000 DCD Section 3.9 discuss aspects of how thermal stratification is addressed in the design. AP1000 fatigue evaluations consider PRHR operation and its effect on RCS piping and components.

   The following will be added to the end of WCAP Section 2.3.5.5:

   "In addition to the above considerations, thermal stratification is considered in the design of the AP1000 piping systems including the RCS loop piping, components, and PRHR piping. Fatigue evaluations consider PRHR operation and its effect on RCS piping and components. DCD section 3.9 discusses aspects of thermal stratification in the AP1000 design."

5. As defined in the AP1000 PRA, Chapter 3, Table 3-2, the AP1000 PRA covers both spurious actuation of ADS-4 valves and spurious actuation of ADS Stage 1, 2 and 3 valves.

   Reference to PRA Chapter 3, Table 3-2 will be added to P. 3-16, Item 2.3.6.4 "ADS actuation during power operation".

6. All typos identified will be corrected except that noted for P. 3-15, 3rd paragraph: "AP600" should be "AP1000". This typo could not be located on P 3-15.

7. The following new sections will be added to the WCAP:

   2.2.14 Diverse Actuation System

The AP1000 Diverse Actuation System (DAS) provides diverse, alternate, manual and automatic actuation of selected engineered safety features. DAS and PMS share actuated devices (i.e., valves). The interface to the actuated devices is designed so that the DAS can not prevent the PMS from actuating any of the shared actuated equipment.

DAS also provides dedicated, independent indication of selected plant parameters. No sensors are shared between DAS and PMS.

The DAS setpoints and tuning constants are calculated to allow the PMS to actuate before DAS.

3.5 Commission Error Due to I&C Failures

The AP1000 defense-in-depth provides redundant and diverse means to monitor and accomplish each safety function, reducing the possibility of unmitigated interactions.

Passive systems have reduced the need for operator response. If an I&C system fails, operators are directed to use backup and/or alternate means to respond, as in ATWS scenarios or a Common Mode Failure analysis. The systems remaining available for use will depend upon the hypothesized failure and its results. However, the reliability of the I&C architecture and the adequacy of its defense-in-depth are established independent of the issue of adverse system interactions. Bounding single failures of plant equipment are addressed by FMEA and safety analysis. Multiple failures of equipment and operators are addressed by PRA and CMF analysis.

The AP1000 uses a computerized procedure system. As stated in DCD Section 18.8, the design of a backup to the computerized procedure system, to handle the unlikely event of a loss of the computerized procedure system, is developed as part of the human system interface design process. Design options include the use of a paper backup. The AP1000 human system design process is described in DCD Chapter 18.

Credible human errors of commission are considered in this report as possible complicating factors for selected adverse system interactions. No credible human errors of commission were found to have significant impact by the present analysis. Therefore, no additional issues were raised about the adequacy of the I&C systems design, including the extent of its backup systems and facilities.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.183

Question:

In section 4.3.2.7.3, Prediction of the Core Stability, the first paragraph makes reference to the fact that AP1000's 14-foot core is slightly less stable with respect to axial xenon oscillations than the 12-foot cores. The same paragraph points to supporting information provided in subsequent section 4.3.2.7.4 for explaining this increase in instability. But this section only provides information regarding 12-foot cores. No data is provided in support of a 14-foot core such as that from South Texas Plant, Units 1 and 2. Please provide additional technical justification in support of the 14-foot core for the increased axial xenon stability.

Westinghouse Response:

For the safety analysis report (SAR) of the South Texas Nuclear Power Plant (Reference 440.183-1), the first 14-foot Westinghouse Pressurized Water Reactor (PWR), the axial power distribution stability was investigated. The analysis was made by the PANDA code. The code accuracy for the axial xenon oscillation analysis was verified in detail against the xenon oscillation tests performed at a Westinghouse PWR as stated in the South Texas SAR Section 4.3.2.7 and Reference 34 of the AP1000 DCD Section 4.3.5. It was concluded that the PANDA code conservatively predicted the stability index with a margin of approximately 0.01 hr^{-1} in the stability index.

Comparison was made for typical 12-foot and 14-foot Westinghouse PWRs. At BOL (150 MWD/MTU), stability indexes of about -0.047 hr^{-1} and -0.020 hr^{-1} were obtained for 12-foot and 14-foot cores, respectively. The axial stability index is essentially zero in the 11,000 to 12,000 MWD/MTU range for 12-foot cores and in the 8,000 to 9,000 MWD/MTU range for 14-foot cores. At extended burnup (more than ~15,000 MWD/MTU) both 12-foot and 14-foot cores have essentially the same stability index of about 0.02 hr^{-1} or less.

The axial oscillation period is comparable for both 12-foot and 14-foot cores. A period of 27 and 28 hours is obtained for the 12-foot and 14-foot cores, respectively, at BOL. At EOL, periods of about 32 and 34 hours were obtained for 12-foot and 14-foot cores, respectively. These values depend on the core design as well as burnup, and the stability index can be positive throughout core life both for 12-foot and 14-foot cores. However, the long period and vertical control rod system make axial xenon transients easily controllable in modern PWRs at all times of life. As stated in Section 4.3.2.7.3 of the AP1000 DCD, free axial xenon oscillations are not allowed to occur for a core of any height, except during special physics tests.

In 1988, South Texas Units 1 and 2 started commercial operation. These plants are the standard Westinghouse PWRs with a 14-foot core. Two other Westinghouse 14-foot PWR plants, Tihange Unit 3 and Doel Unit 4, have been operating in Belgium since 1985. Many more

Westinghouse type 14-foot plants are operating in France. No uncontrollable axial xenon oscillation incident has ever been reported either in 12-foot or 14-foot cores of the Westinghouse type nuclear power reactors.

Reference:

440-183-1: Section 4.3-9 of South Texas Project Electric Generating Station Unit 1 & 2 UFSAR

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.184

Question:

New Generic Issue 163, "Multiple Steam Generator [SG] Tube Leakage," in NUREG-0933, "A Prioritization of Generic Safety Issues," identifies a safety concern associated with potential multiple steam generator tube leaks triggered by a main steam line break outside containment that cannot be isolated. This sequence of events could lead to core damage due to the loss of all primary system coolant and safety injection fluid in the refueling water storage tank. The Nuclear Regulatory Commission (NRC) has given this issue HIGH priority ranking, and is working toward a resolution of the issue. The AP1000 design control document (DCD) Section 1.9, "Compliance with Regulatory Criteria," does not address this issue, except that Table 1.9-2 indicates that Generic Issue 163 is unresolved pending generic resolution.

(A) Please provide an evaluation of the AP1000 design with respect to coping with the safety concern of Issue 163 regarding multiple SG tube leakage resulting from a steam line break outside containment.

(B) Please discuss how any subsequent requirements that may be imposed by the NRC as a resolution of this issue will be identified to a prospective combined license applicant that references the AP1000 design, i.e., how are we given assurance that a combined license applicant will commit to complying with any requirements that may be imposed by the NRC as a resolution of Issue 163.

Westinghouse Response:

A. The AP1000 plant response to a main steam line break (MSLB) scrams the reactor automatically and removes decay heat via the intact generator or the passive RHR heat exchanger. If the MSLB is not isolated the RCS will continue to lose coolant after shutdown through leaking steam generator tubes, the plant responds to the scenario as a small loss of coolant accident. The core makeup tanks drain and produce a low level signal. The plant protection and monitoring system depressurizes the RCS via the automatic depressurization system (ADS). The core remains covered throughout the scenario. Once the RCS is depressurized, the much lower containment pressure stops the containment water loss through the leaking steam generator tubes. Therefore, no long-term core uncovery is expected.

B. Based on the above discussion, this issue should be considered closed for the AP1000, and no additional action item for the Combined Licensing Applicant should be required in the DCD.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.043 (Response Revision 1)

Question:

WCAP-15847, page 2-1. Reference 1 should be revised if it is to remain consistent with current NRC guidance.

Westinghouse Response:

WCAP-15847, page 2-1 will be revised, as indicated below.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

From WCAP-15847, page 1-1:

1 INTRODUCTION

Chapter 18 of the AP1000 Design Control Document (DCD) contains the AP1000 Design Certification information for Human Factors Engineering ~~against NUREG-0711 (Reference 1)~~.

From WCAP-15847, page 2-1:

2.0 REFERENCES

1. Reference Deleted. ~~NUREG-0711, Human Factors Engineering Program Review Model, July 1994.~~
2. WCAP-12601 Revision 19, Westinghouse AP600 Program Operating Procedures Document.

NRC Additional Comments:

Reference: WCAP-15847, "AP1000 Quality Assurance Procedures Supporting NRC Review of AP1000 DCD Sections 18.2 and 18.8," Section AP-3.14, "Plant Instrumentation and Control [I&C] System."

With regard to instrumentation and control design process, WCAP-15847 has not addressed the Life Cycle Design Process for a digital I&C protection system. Westinghouse should revise WCAP-15847 to include the Life Cycle Design Process as discussed in WCAP-15927, "Design Process for AP1000 Common Q Safety Systems," and WCAP-13383, "AP600 Instrumentation and Control Hardware and Software Design, Verification, and Validation Process Report."

Westinghouse Additional Response:

Westinghouse agrees that WCAP-15847 does not address the Life Cycle Design Process. WCAP-15847, "AP1000 Quality Assurance Procedures Supporting NRC Review of AP1000 DCD Sections 18.2 and 18.8," only includes those procedures to be used during and in support of the Human System Interface (HSI) design; therefore, the complete Life Cycle Design Process does not need to be included in WCAP-15847.

WCAP-15927 and WCAP-13383 are provided to describe the complete I&C design and development process in support of DCD Chapter 7. As noted in the NRC additional comments, the Life Cycle Design Process is included in these WCAPs.

No change to WCAP-15847 is needed.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 650.001

Question:

In Section 6.3.2.2.7.2 of the AP1000 Design Control Document (DCD), entitled IRWST [in-containment refueling water storage tank] Screens, and Section 6.3.2.2.7.3, entitled Containment Recirculation Screens, the applicant states that the clearance of the IRWST and containment recirculation screens prevents debris larger than 0.125 inches from infiltrating the reactor coolant system and blocking fuel cooling passages. However, in Section 3.4.1.2.2.1, entitled Containment Flooding Events, the DCD also states that, following a loss-of-coolant accident (LOCA), the water level in containment would be sufficiently high to provide water flow back into the reactor coolant system via the break location. . . .

A breached reactor coolant system pipe would apparently present an unfiltered pathway into the primary system, through which pieces of debris orders of magnitude larger than 0.125 inches could infiltrate, because the IRWST and containment recirculation debris screens would be bypassed. Although the AP1000's safety-related core-cooling flowpaths do not contain the typical flow restrictions that have been considered for operating plants (e.g., pump clearances, spray nozzles, and throttle valves), the debris filters on the fuel assembly bottom nozzles appear (based upon the NRC staff's review of Section 4.2.2 of the DCD, and accompanying figures) to present a potentially adverse debris accumulation point. Therefore, the Nuclear Regulatory Commission (NRC) staff requests additional information from the applicant to ensure that the AP1000 design adequately considers the potential for debris to infiltrate the reactor coolant system through an unfiltered pathway (e.g., a ruptured pipe) and interrupt reactor core cooling by blocking requisite flowpaths.

Westinghouse Response:

For a postulated breach in the AP1000 reactor coolant system pipe, sufficient liquid would be provided from the IRWST to raise the level of the inventory in containment above the piping break location. For a postulated double ended guillotine break, at least 3600 seconds (1 hour) is required to raise the water level on the containment floor to above the reactor coolant system piping.

By the time the pool elevation reaches the break location, it is relatively quiescent. Blowdown will have been completed. IRWST drain down will be ongoing with the driving head of the flow from the IRWST to the reactor approaching its long-term cooling equilibrium value. Rooms and corridors below the flood level will have been filled. In general, other than around the break location, fluid velocities and turbulence levels in the pool will be low.

A source of debris having a size of 0.125 inches or larger is the reflective metallic insulation (RMI) on high energy piping runs. The postulated piping break would result in the destruction of RMI in the immediate vicinity of the break location into debris of various sizes. RMI, however, is dense compared to water and will settle to the containment floor as the pool builds. The long time taken by the pool to reach the postulated break location provides assurance that this debris will settle well before the water level reaches the break location.

Furthermore, for a postulated hot leg break, all flow from the IRWST would necessarily flow through the reactor vessel and out the break. This flow path would preclude the ingress of debris into the RCS.

For a postulated cold leg break, coolant flow to the core is equal to the boil-off rate needed to remove decay heat. Early in the event, with the containment water level below the break location, excess IRWST flow is ducted out the break. After the containment water level rises above the break elevation, it is possible for water to flow into the RCS, although it is unlikely that particulate debris will still be in suspension at this time, at least one hour after the accident. Therefore, it is also unlikely that debris would be ingested into the RCS through the postulated cold leg break.

Although it is unlikely, should debris larger than 0.125 inches be ingested through the postulated cold leg break location, it would settle in the bottom of the reactor vessel. As also noted in the preceding paragraph, for a postulated cold leg break, flow to the core would match boil-off. It has been previously demonstrated that post-accident boil-off flow rates in a typical large PWR (approximately 3400 mWt) are insufficient to transport 0.04-inch debris having a specific gravity of about 1.3 into the reactor core (Reference 1). This conclusion is applicable to the AP1000.

Thus there is no suitable transport mechanism to cause a blockage of the fuel assembly inlets. But even if local fuel inlet blockages were postulated to occur at the time when the water level in containment rises above the break, the open lattice structure of the fuel design will allow for significant cross-flow and mixing of coolant in the core. This cross-flow and mixing would provide sufficient core cooling.

In summary, debris of the size to be a concern is generated by the destruction of RMI by the break flow. Because of the high density of the RMI debris, the long time required to build the level of liquid in the pool to the break location, and relative quiescence of the pool, such debris will have settled to the containment floor prior to the liquid level reaching the break location. In the unlikely event that debris were suspended pool, ingress of debris into the AP1000 RCS by way of a postulated hot leg break would be precluded as all flow from the IRWST is ducted through the core and out that break location. Similarly, for a postulated cold leg break, IRWST flow in excess of the boil-off rate is exhausted through the break location. Should debris find its way into the RCS through a postulated cold leg break, the fluid velocities in the lower plenum of the reactor vessel are sufficiently low as to provide for the debris to settle in that volume.

Therefore, although the flood-up level of liquid in the AP1000 containment is above the RCS piping, postulated breaks in that piping do not provide a path for debris larger than 0.125 inches to be ingested into the RCS.

References

1. Andreychek, T. S., "Evaluating Effects of Debris Transport within a PWR Reactor Coolant System during Operation in the Recirculation Mode," Proceedings of the 4th Miami International Symposium on Multiphase Transport and Particulate Phenomena, 1986

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 650.002

Question:

In Section 6.3.2.2.7.1, entitled General Screen Design Criteria, the DCD states that reflective metallic insulation is used on ASME [American Society of Mechanical Engineers Code] class 1 piping lines because they are subject to loss-of-coolant accidents. Additionally, the DCD states that the potential targets of jet impingement from analyzed LOCA pipe breaks, including the reactor vessel, reactor coolant pumps, steam generators, pressurizer, and unshielded piping lines, are also insulated with reflective metallic insulation or an equivalent type of insulation. The DCD then concludes that, [a]s a result, fibrous debris is not generated by loss-of-coolant accidents.

On the basis of research and analysis undertaken to support the NRC staff's efforts in resolving Generic Safety Issue 191 (GSI-191), Assessment of Debris Accumulation on PWR Sump Performance, the staff questions the validity of the conclusion that fibrous debris will not be generated by a LOCA at an AP1000 reactor. Specifically, the NRC staff and licensees of currently operating plants have identified that the loose dispersion of dust and dirt that resides on the surfaces of the components and structures within the containment can consist of significant amounts of fibrous material, even at plants that do not deploy fibrous insulation in the zones of insulation destruction for postulated pipe ruptures. It is thought that the constituents of this resident fibrous material originate from such items as cloth protective clothing and equipment covers, human hair, fines from fibrous material (e.g., thermal insulation and fire barriers) outside of destruction zones, and sources external to the containment (when the equipment hatch or other containment apertures are open). This fine, dispersed resident fibrous material may be washed down toward the IRWST or containment recirculation screens by break flows, condensate droplets, or other containment drainage flows, and its fineness allows it to remain in suspension for extended periods in a pool of water, even at low turbulence conditions.

Operating experience at boiling-water reactors (BWRs) and NRC-sponsored research indicate that thin fibrous debris beds (i.e., as thin as 1/8 inch in thickness) are capable of filtering a significant fraction of influent particulate debris, which would lead to substantial increases in head loss if high particulate to fiber mass ratios were reached (Reference NUREG/CR-6762, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," Volume 1, dated August 2002, and NUREG/CR-6367, "Experimental Study of Head Loss and Filtration for LOCA Debris," dated February 1996, etc.). To cover a surface area equal to that of both the two IRWST screens and the two containment recirculation screens with a 1/8-inch-thick debris bed, an available volume of resident fibrous material of less than 3 cubic feet would be required. Conventional cleanliness programs notwithstanding, the NRC staff considers it improbable that even the cleanest of operating plant containments would contain an available quantity of resident fibrous material less than 3 cubic feet. Based upon the staff's review of the DCD, it does not appear that the AP1000 containment cleanliness programs is substantially different than the current industry standard, and, therefore, available quantities of resident fibrous material greater than 3 cubic feet existing in containment would also seem to be credible for the AP1000.

On the basis of the above observations, the NRC staff requests further information to determine whether the AP1000's IRWST and recirculation screen designs adequately account for the potential concern related to resident fibrous material, and also, resident particulate matter, such as dirt, which would seem inevitably to be present on containment surfaces.

Westinghouse Response:

See the response to RAI 650.005, on containment recirculation screens and 650.004, on IRWST screens. These responses show the results of calculations of pressure drops across these screens with conservative amounts of resident debris assumed to have been transported to the screens.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 650.003

Question:

Although the NRC's GSI-191 research program has indicated that fire barriers consisting of fibrous material may generally contribute a smaller volume of LOCA-generated debris than fibrous insulation materials, for the AP1000 (which does not employ fibrous insulation in destruction zones), fire barriers could conceivably contribute a significant fraction of the overall quantity of fibrous material generated by a LOCA. In Section 9.5.1.2.1.1, entitled Plant Fire Prevention and Control Features, the DCD states that [c]omplete fire barrier separation necessary to define a fire area is not provided throughout the primary containment fire area...,and that [s]elected cables of a safety-related division which pass through a fire zone of an unrelated division are protected by fire barriers. The staff could not determine from the DCD (a) whether the fire barriers referred to in Section 9.5.1.2.1.1 would consist of fibrous material, and (b) whether these fire barriers would be located in a zone of destruction for a postulated pipe rupture. Please provide this additional information.

Westinghouse Response:

(a) The AP1000 fire barriers in containment are made of steel plates or of "Durasystem" barriers or equivalent. "Durasystem" barriers are composite panels of fiber cement mechanically bonded to punched steel plates on both outer surfaces. Fibrous materials in the panels are bonded to and within Portland cement and any pieces of this material will be captured by the outer steel plates or be sufficiently dense to sink rapidly in water.

(b) DCD Tier 2 Section 9A.3.1.1.8 describes the fire barrier separation necessary for the selected cables of safety-related divisions that pass through a fire zone of an unrelated division. In all cases they are described as barriers of steel or steel-composite materials. Thus, there is no fibrous fire barrier material located in a zone of destruction for a postulated pipe rupture.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 650.004

Question:

Section 6.3.2.2.3 of the DCD, entitled In-Containment Refueling Water Storage Tank, states that [t]he IRWST is stainless steel lined and does not contain material either in the tank or the recirculation path that could plug the outlet screens. Though the water in the IRWST would likely be relatively pure, the staff believes it is not likely to be completely free of debris, particularly considering the debris-concentrating potential afforded by the cycling of the IRWST inventory during refueling outages, and the opportunity for any suspended debris to settle to the IRWST floor during long periods of stagnation. The staff notes that as little as 1.5 cubic feet of fibrous debris could potentially cover both IRWST screens, which is a very small fraction of the tank's capacity of 73,900 cubic feet. If an automatic depressurization system (ADS) actuation occurs during an accident condition, any debris residing on the bottom of the tank (including heavier particulate matter) could be easily re-suspended by the consequent induced turbulence. Considering these NRC staff observations, please provide further information to clarify why fibrous and particulate debris settling onto the tank floor is not a concern for the IRWST screens, and clarify that the analysis concerning debris transport and head loss provided accounts for the most limiting IRWST conditions (e.g., during potentially turbulent conditions and at reduced tank levels as the switchover to recirculation approaches).

Westinghouse Response:

The IRWST is a stainless steel lined storage tank located inside containment. Refer to Figure 650.004-1 and -2. The tank is normally closed off from the containment although there are large louvered vents located through the IRWST roof. The tank is 30.25 feet high (inside floor to ceiling). The tank extends around part of the containment from the refueling cavity past the pressurizer and is about 120 feet across. There are two ADS spargers located in one side of the tank. The bottom of the sparger arms are located about 16 feet above the floor of the tank. There are two separate PXS injection line connections from the bottom of the IRWST. Line A is on the ADS side and line B is on the other side with the PRHR HX. Each of these lines is protected by a screen assembly. Line A has a connection to the Spent Fuel Pool Cooling System (SFS) pumps. This line allows the SFS to recirculate the water in the IRWST to provide cooling and purification as desired; the SFS recirculation flow returns to the IRWST in the PRHR HX side of the tank. This SFS can also be used to transfer IRWST water to and from the refueling cavity. Line B has a connection to the Normal Residual Heat Removal System (RNS). This line allows the RNS to recirculate the IRWST for purposes of mixing and cooling. Line B can be used to allow the RNS to provide long term post ADS core cooling via containment recirculation. The RNS does not use this line to provide short term low pressure injection; the RNS can provide this function using the SFS cask loading pit as a source of water.

Cleanliness of the IRWST Water

The IRWST is expected to be clean because of the purification provided by the SFS and the fact that this water is used during every refueling; if the water was not clean it would cause delays in the refueling operations. The SFS is used to transfer the IRWST water to and from the refueling cavity. The SFS also provides purification of the refueling cavity while the IRWST water is in the refueling cavity. The SFS has a demineralizer / filter connected with each SFS pump that can purify a significant portion of the total SFS pump flow. This purification is used when the IRWST is transferred to the refueling cavity, during refueling operations and during the transfer back to the IRWST. In addition, the SFS is expected to be used to purify the IRWST in a recirculation mode for several days following refueling operations. These operations should eliminate resident debris from being suspended in the IRWST water at the start of an accident and to limit the amount of resident debris that might settle out on the IRWST floor.

Potential for Stirring Up Debris Lying on IRWST Floor

As discussed above significant quantities of debris are not expected to settle out in the IRWST during normal operations. Even if such debris were to have settled onto the IRWST floor, it is not expected to be stirred up in the limiting accident. The ADS valves are designed to control the depressurization rate of the RCS. These same characteristics also limit the peak mass flow into the IRWST.

- ADS stage 1 are 4" valves. ADS stage 2/3 are 8" valves.
- ADS stage 1 valves open in 20 to 30 sec.

In a DVI LOCA, the ADS 1,2,3 valves only pass significant flow for about 6 minutes between their opening and shortly after the opening of the ADS 4 valves, as shown on DCD figure 15.6.5.4CB-67. During this time the IRWST remains highly subcooled, which significantly reduces the agitation of the IRWST. Note that the bottom of the ADS sparger arms is 16 ft above the bottom of the tank. Once the ADS 4 valves open the flow through the ADS 1,2,3 valves drops to very low levels. A DVI LOCA is limiting with respect to IRWST injection performance because only one line is able to provide injection to the RCS while the other line spills. Note that the ADS spargers are located on one side of the IRWST. Any resident debris that might have settled out on the other side (PRHR HX side) will not see any significant agitation. Early in a DVI LOCA the flow injection from the IRWST through the intact DVI line is less than 180 lb/sec. This flow results in only 0.04 ft/sec flows at the face of the IRWST screens faces. Throughout the rest of the IRWST there is essentially no flow. As a result, any resident debris that might have settled on the IRWST floor prior to an accident is not likely to be stirred up by the ADS and any particles that are heavier than water will settle out before they can be transported to the screens.

In non-DVI LOCAs there are several factors that make the event less limiting:

- There will be injection flow through both DVI lines to the RCS.
- ADS will be actuated later with lower decay heat levels.
- The injection flows will be lower per DVI line / IRWST screen.

Potential for Washing Debris into IRWST During a Accident

It is unlikely that significant amounts of debris will be swept into the IRWST during an accident. The only process that is available to wash resident debris into the IRWST is the steam that condenses on the inside of the containment shell. These surfaces include the vertical containment walls and the containment dome. Although these surfaces are large, their orientation and their location well above the operating deck limit their potential residual debris loading. Although the AP1000 has a limited containment spray capability, it is only intended to be used following a core melt accident. As a result, the AP1000 will not see the large containment spray flows that greatly increase the amount of containment surfaces that would be washed into the IRWST.

Ability of IRWST Screens to Tolerate Debris

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

The assumptions used in the analysis include:

- A total of 500 lb of resident debris is assumed to be available in the containment. It is assumed that this debris is divided 50/50 between fibrous and particles. Further more, it is conservatively assumed that all of this debris is transported to the IRWST screens. Note that even with a DVI line break and a single failure there will be flow through both IRWST screens. These conservative assumptions lead to 2.0 ft3 of fibers deposited on each of the 2 IRWST screens. Refer to RAI response to 720.005-R1 for a more detailed discussion of the debris.
- With a screen area of 70 ft2 each, the resulting fibrous bed thickness will be 0.34 inches.
- The flow rate through the intact DVI line is assumed to be 160 lb/sec. This flow is based on the peak IRWST injection flow through the intact DVI line which occurs early in the accident (about 2700 sec, DCD figure 15.6.5.4B-71). At the IRWST water conditions for this event, the volumetric flow rate would be 1170 gpm.
- At this flow rate, the screen face velocity with this flow is 0.037 ft/sec.
- With the above amounts of debris and flow rates, the pressure loss across the debris is less than 0.09 feet water or 0.037 psi. The basis for the pressure loss is presented in RAI 720.005.

- The pressure loss through the intact IRWST injection line from the IRWST to the RV downcomer is more than 5.8 psi at this flow rate. The increase in screen DP shown above is only 0.6%. The IRWST injection flow would only have to decrease 0.3% to compensate for this increase in screen DP. The bounding pressure loss through the resident debris is insignificant relative to the pressure loss in the line.

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Figure 650.004-1 AP1000 IRWST Plan View

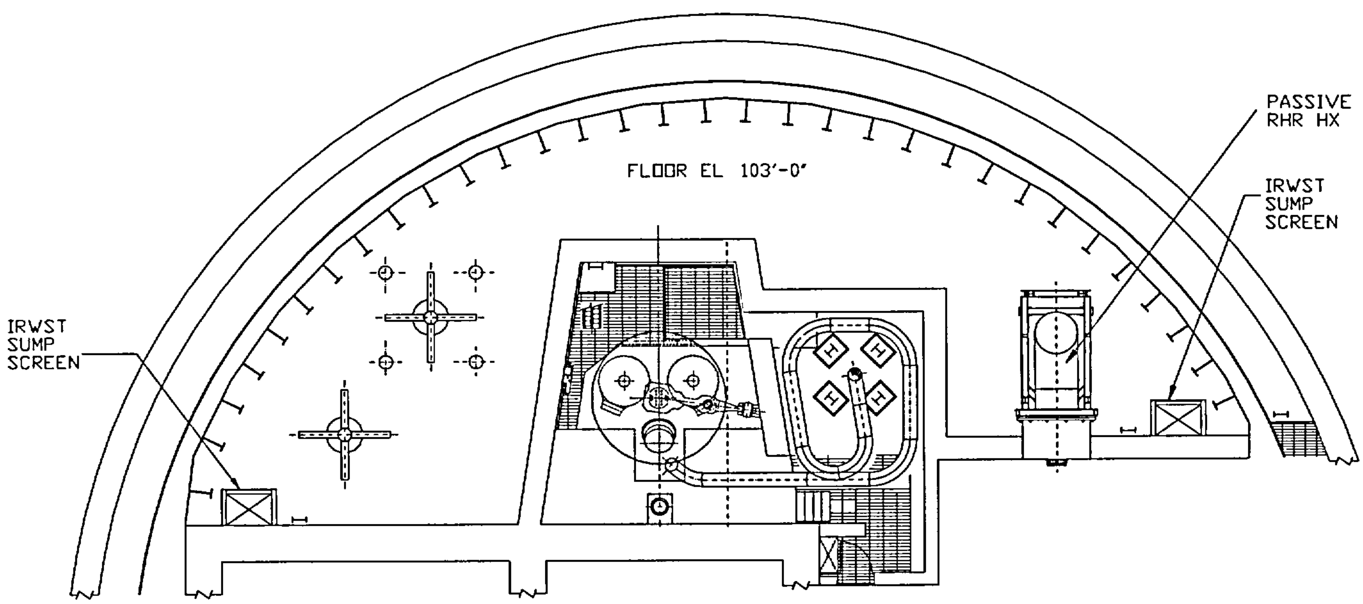
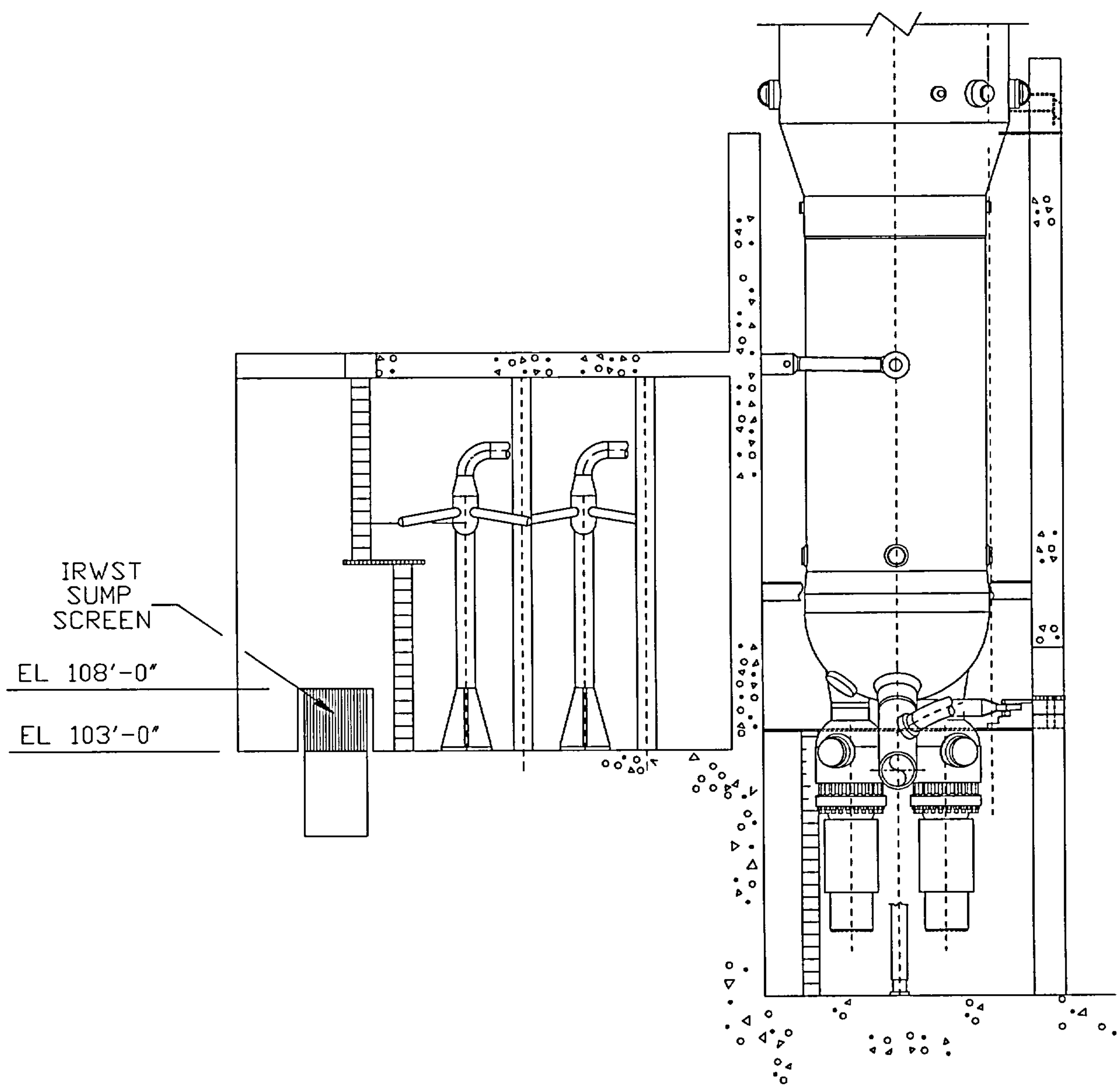


Figure 650.004-2 AP1000 IRWST Section View



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 650.005

Question:

Based upon its review of the DCD, the NRC staff was unable to locate an analysis of the debris-blockage failure criteria of the IRWST and recirculation screens. This apparent omission may be due to the fact that the applicant considers debris blockage failures to be incredible for design-basis events based upon the debris source control measures specified in the DCD. However, based upon the NRC staff's concerns related to resident fibrous material and other potential sources of fibrous debris (reference items 2 through 4 above), the staff believes it is possible that quantities of fibrous debris capable of blocking the entire surface areas of the IRWST and recirculation screens could be generated by a LOCA at an AP1000. As such, the staff believes it is essential for the applicant to provide further detail concerning: (a) how large a pressure head is available from natural circulation to drive the required flow rates through the IRWST and recirculation screens, (b) the maximum postulated head loss across the IRWST and recirculation screens, and (c) how much margin exists between the values for items (a) and (b).

Westinghouse Response:

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

The postulated DVI line break is taken as the limiting break for responding to this issue. A DVI line break in a PXS room may render valves in the associated recirculating line inoperable because it can flood the recirculation squib valves before they are actuated. As a result, all of the recirculation flow will pass through a single recirculation screen. This condition would maximize the potential for debris transport to and collection on the single operating recirculation screen, and head loss across the resulting debris bed. In addition, a DVI break in one of the PXS rooms results in lower containment flood levels. For LOCAs in other locations, both recirculation screens will be available which will result in lower flow rates through each screen. In addition, the containment flood levels will be higher.

Under long term cooling for a postulated DVI line break, the flow rate through the single operating recirculation screen is estimated to be about 180 lb/sec (refer to DCD figures 15.6.5.4C-13 and -14). This translates to a velocity of about 0.04 ft/sec at the recirculation screen face. This velocity would be smaller in other areas of the containment that are removed from the containment recirculation screens.

The parametric sump blockage evaluation performed for GSI-191 and reported in LA-UR-01-4083 (Reference 1) assumed a range of 100 lb to 500 lb for latent containment debris. The larger value will be used for this evaluation. Further, it will be assumed that 50% or one half of the latent containment debris will be in the form of fiber. Thus, the latent containment loading of fiber for this evaluation is assumed to range from 50 to 250 lb. Further, invoking the assumption that the fiber is buoyantly neutral, the volume of latent fibrous debris inside containment is calculated as:

$$debris\ volume = {}^{debris\ mass}\!/_{debris\ density}$$

Using the assumption of neutral buoyancy:

$$debris\ density = 62.4\ lb/ft^3$$

The maximum fibrous debris is calculated to be:

$$Volume = 4.0\ ft^3$$

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

$$thickness = 0.057\ ft = 0.69\ inches$$

Thus, assuming a latent containment debris loading of 500 lb, 50% of which is assumed to be fibrous, and assuming all of the latent fibrous debris is deposited on to the single operating recirculation screen, the resulting debris bed thickness is conservatively calculated to be approximately 5/8 inches thick.

There are several conservative assumptions incorporated into the fibrous debris bed thickness calculated above.

- First, a bounding value of 500 lb of latent containment debris was used. Although this is consistent with the parametric study performed for GSI-191 and the operating fleet of PWR's, it is more than three times larger that the values used in addressing BWR strainer blockage.
- Second, the volume of the available fiber is maximized by assuming the fiber density is equivalent to water. Use of a larger value of the density for latent fiber debris results in a smaller volume of the debris which, in turn, results in a lower fibrous debris bed thickness.
- Finally, although it is recognized that even at low flow rates, buoyant-neutral debris may migrate to the recirculation screen, no credit was taken for settling. From Table B-3, "Fibrous Debris Classification by Shape," of NUREG/CR 6224 (Reference 2), it is noted that even single strands of fiber will settle in calm pools. This evaluation took no credit for settling of fibrous debris.

From Figure B-19 of NUREG/CR-6224 (Reference 2), assuming a flow velocity of 0.04 ft/sec, the head loss through a pure fiber bed on the recirculation screen is taken to be:

$$\Delta P/\Delta t \quad = \quad 0.25\ ft_{H20}\ /\ inch\ of\ \ debris\ bed\ thickness$$

For the thickness of the debris bed calculated, the total pressure drop through a pure fiber debris bed calculated for the AP1000 containment recirculation screen is:

$$\Delta P_{FIBER} \quad = \quad 0.25\ ft_{H2O} \times 0.69\ inches \quad = \quad 0.18\ ft_{H2O} \quad = \quad 0.075\ psi$$

For the purposes of this evaluation, the following assumptions are made regarding latent containment debris:

- 50% of the debris mass is fiber, and the remaining 50% is particulate debris, and,
- All particulate debris is deposited on the fibrous bed, the particulate-to-fiber mass ratio on the debris bed would be 1 (250 lb of latent fiber debris and 250 lb of latent particulate debris).

From Figure B-24, "Comparison of NRC Head Loss Experiments Data with the NUREG/CR-6224 Correlation," the pressure drop multiplier for converting the pressure drop for a pure fiber bed to that expected to result from a mixed bed is observed to be a value of 2. Therefore, the expected pressure drop through a mixed fiber / particulate debris bed for the AP1000 containment recirculation screen is evaluated to be:

$$\Delta P_{MAX} \quad = \quad \Delta P_{MIXED}\ /\ \Delta P_{FIBER} \quad \times \quad \Delta P_{FIBER}$$

$$\Delta P_{MAX} \quad = \quad 2 \quad \times \quad 0.075\ psi \quad = \quad 0.15\ psi$$

This increase in pressure drop through the recirculation screen is insignificant (~5%) compared with the 2.8 psi in these lines during the DCD analysis. Note that the recirculation flow would only have to decrease ~ 2.5% to compensate for this increase in screen DP. The assumption of a 50 / 50 split between latent fibrous and particulate debris is considered a reasonable engineering judgement. Greater fiber will increase the depth of the fiber bed, but reduce the pressure drop multiplier associated with particulate debris. Lesser amounts of fiber will increase the pressure drop multiplier associated with particulate debris but reduce the fiber bed and the corresponding pressure drop through that bed.

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

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

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

- The maximum debris loading on the containment recirculation screen is assumed. No credit is taken for the holdup of latent containment debris elsewhere in the containment (in dead-ended cubicles and rooms, on IRWST screens, etc.)

- A conservatively low density for the latent fibrous debris was assumed. Assuming the latent fibrous debris had a density equal to that of water provided for a maximum volume of fibrous debris, and hence a maximum thickness of the resulting debris bed, on the recirculation screen.

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

References:

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

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 650.006

Question:

During the NRC staff's review of the AP600, an open item identified as OITS #6590 was generated. Westinghouse responded to this open item in three parts. Please confirm whether or not the second and third parts of the response to OITS #6590 also apply to the AP1000 design. Specifically, (a) would the complete failure of the non-safety-related coatings block any portion of the AP1000 recirculation screens? and (b) is a combined license (COL) action required for the AP1000 that an analysis must be performed of coating debris generation and transport that is based upon appropriate test data?

Westinghouse Response:

The response to AP600 Open Item # 6590 was originally transmitted to the NRC in Westinghouse letter DCP/NRC1251 dated February 10, 1998. The response was later revised in Westinghouse letter DCP/NRC1271 dated February 27, 1998. The RAI above is referring to the original response that was answered in three parts. The revised response to OITS #6590 was answered in four parts. The following response to items (a) and (b) above reflect the revised response to OITS#6590:

(a) In DCP/NRC1271, results of calculations are presented that demonstrate that failure of nonsafety-related coatings inside containment would not block any portion of the AP600 containment recirculation screens. That response has been updated for AP1000 and is presented below.

The AP1000 has several unique characteristics that allow the plant to tolerate the failure of nonsafety-related coating used inside containment. Table 650.006-1 attached to this response provides a list of these characteristics. These characteristics include long settling times between the end of RCS blowdown during a LOCA and the beginning of recirculation; the large water volumes provided by the passive core cooling system (PXS) and the shape of the containment lower volumes provides for high flood-up levels. These high flood-up levels allow the containment recirculation screens to be located relatively high. The bottoms of the screens are located well above the lowest elevations of the containment and this allows coating debris to settle out without challenging the bottom of the screens. The screens are very tall, which further reduces the chance that coating debris can reach the screens. The AP1000 screens also have a unique feature (protective screen plates) that have been added to specifically prevent coating debris from entering the post accident containment water close to the screens and potentially blocking the screens. These screen plates are located above each recirculation screen and extend well out in front and to the sides of the screens.

Another AP1000 characteristic that reduces the potential for coating debris blocking the recirculation screens is that fibrous insulation is not used where it may be damaged by a LOCA. This eliminates the potential adverse interaction where fibrous debris acts like a fine filter and collects small particles of dust or coating debris and leads to high screen pressure drops.

As discussed in the AP1000 DCD section 6.1.3.2, the nonsafety-related (service level II) coating material used in the containment will be procured with 10 CFR Part 50, Appendix B quality assurance requirements. As a result, the nonsafety-related coatings are not expected to fail. However, to provide a robust design, the failure of the nonsafety-related coatings is considered.

The application of 10 CFR Part 50, Appendix B quality assurance to the manufacture and procurement of nonsafety-related coating materials allows the characteristics of the paint to be identified in terms of density and failure mechanisms. This information makes it possible to bound the size and density of the debris that might be generated by failure of these coatings and have the potential for blocking the screens. Table 650.006-2 shows the key inputs used in evaluating the coating settling.

Figures 650.006-1 and -2 show the settling trajectories of coating debris starting at the edge of the protective plate. Figure 650.006-1 shows the approach from the front of the screen and figure 650.006-2 shows the approach from the side. Such debris can not enter the water any closer to the screens because no coatings are permitted any closer to the screens. The figure shows that with design settling rates the debris will settle to the elevation of the bottom of the screen after drifting about 6 feet, which is at about 4 feet away from the screen. Significant conservatism was included in this evaluation, including assuming the lightest/smallest coating debris. It also assumes the maximum recirculation flow of 1600 gpm, consistent with the maximum flow from two RNS pump operating unthrottled with suction taken from two screens. Another area where significant margin has been applied is in the debris settling rates. AP1000 uses debris settling rates that contain a factor of two margin compared to the reference settling data (Reference 1) which is sufficient to account for uncertainties. Without this added margin, the coating debris settles out very quickly in less than 3 feet, as shown in Figure 650.006-1 and -2.

Several sensitivity studies have been performed to demonstrate the robustness of the AP1000 design with respect to uncertainty in coating debris settling. The margin applied to settling rates was increased from the AP1000 design value of a factor of 2 to a factor of 4.3 until the debris would block 50% of the screen and to a factor of 14 before debris could block 90% of the screen. Note that the potential screen blockage is based on averaging the results from Figures 650.006-1 and -2. Even with such extreme margins, the recirculation would continue to function because the recirculation screen can tolerate significant blockage and still support recirculation operation. Note that if screen blockage ever reached the point where the RNS pumps cavitated and stopped operating, the PXS would revert to gravity recirculation at its lower flow rate (< 1330 gpm).

The debris settling shown in figures 650.006-1 and -2 provide confidence that the plant can sustain complete failure of the nonsafety-related coatings located inside containment without excessive blockage of the recirculation screens.

(b) In DCP/NRC1271, Westinghouse transmitted the revised response to OITS#6590. The revised response replaced the previous commitment for the combined license applicant (COL) to perform an analysis of coating debris generation and transport based upon appropriate test data with an ITAAC commitment to verify that the nonsafety-related coatings used in the AP600 inside containment were consistent with the coating debris assumptions used in the calculations described in Part (a) of this response. This ITAAC commitment has been retained for the AP1000, and is included in AP1000 Tier 1 Information, Section 2.2.3, Item 8.c) (x) in Table 2.2.3-4.

Reference:

1. Gibbs and Hill Report, "Evaluation of Paint and Insulation Debris Effects on Containment Emergency Sump Performance", Revision 1, September 1994.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Table 650.006-1
AP1000 Post LOCA Recirculation Conditions

Time of Initiation of Recirculation		
- Large LOCA	~5 hr	
- DE DVI LOCA	~5 hr	
Flood up level (water level)		
- above RV cavity floor	36.5 ft	
- above loop compartment floor	25 ft	
Screen elevation	Corridor	Loop
- Protective plate above top screen	< 1 ft	< 1 ft
- Height screen	13 ft	10.2 ft
- Bottom screen above nearby floor	2 ft	2 ft
- Bottom screen above RV cavity floor	15 ft	12.2 ft
Containment Recirculation flow rates		
- Maximum total (no failures, all pumps, both screens)	2600 gpm	
- Expected total (no failures, all pumps, both screens)	2000 gpm	
- Maximum per screen (RNS, all pumps, both screens) (1)	1600 gpm	
- Maximum per screen (PXS, no failure, one screen)	1330 gpm	

Notes:
(1) Adds arbitrary margin to case shown with both RNS pumps and both screens.

Table 650.006-2
Inputs to Coating Debris Settling Calculation

Coating debris:	
- Shape	circular disk
- Diameter	≥ 200 mils
- Thickness	≥ 5 mils
- Density	≥ 100 lb/ft3
Containment Recirculation Screen geometry: (1)	
- Height	13 feet
- Width	5.5 feet
- Distance off floor	2 feet
Screen Protective plate:	
- Height above top screen	1 feet (2)
- Distance plate extends out in front	10 feet
- Distance plate extends out to side	7 feet
Screen flow rate:	
- Maximum flow per screen	1600 gpm (3)

Notes:

(1) The screen located in the corridor is used in this study because it is more limiting than the screen in the loop compartment; it is more limiting because it is taller than the screen in the loop compartment and it has the same plate dimensions.

(2) The AP1000 has an ITAAC limit of 1 foot. The actual design is 0 feet.

(3) The maximum flow per screen is based on operation of the RNS following ADS operation; PXS operation results in lower screen flows (< 1330 gpm). The following conservative assumptions are made in calculating the maximum RNS pump driven flow rates. Two RNS pumps are assumed to take suction from two recirculation screens. The RCS pressure is assumed to be equal to the containment pressure. The RNS pump is assumed to have a conservatively high head vs flow characteristics. Cavitation of the pump due to inadequate NPSHa is conservatively ignored. The piping and equipment flow resistances are assumed to be low.

Figure 650.006-1 AP1000 Coating Debris Settling (Front)

This figure shows that the AP1000 has significant margin to accommodate uncertainty in coating settling rates. This figure shows the approach to the front of the containment recirc screen located in the corridor. Sensitivity of coating settling to debris settling rates is shown. The heavy solid line shows the AP1000 design case, which includes a margin factor of 2.00 times the reference settling data (Reference 1). Lighter lines represent sensitivity studies with greater margins. The margins applied were chosen to force the debris to settle out covering half of the screen (factor of 4.3) and covering 90% of the screen (factor of 14). Even with 90% of the screen completely blocked, either the RNS or the PXS would still be able to provide core cooling. A light line shows the settling using the reference settling data. Note that the potential screen blockage is assumed to be an average of figure 650.006-1 and -2. All of these cases assume a conservatively high flow which bounds the maximum RNS flow possible assuming both RNS pumps take suction from two screens and is unthrottled (1600 gpm/screen).

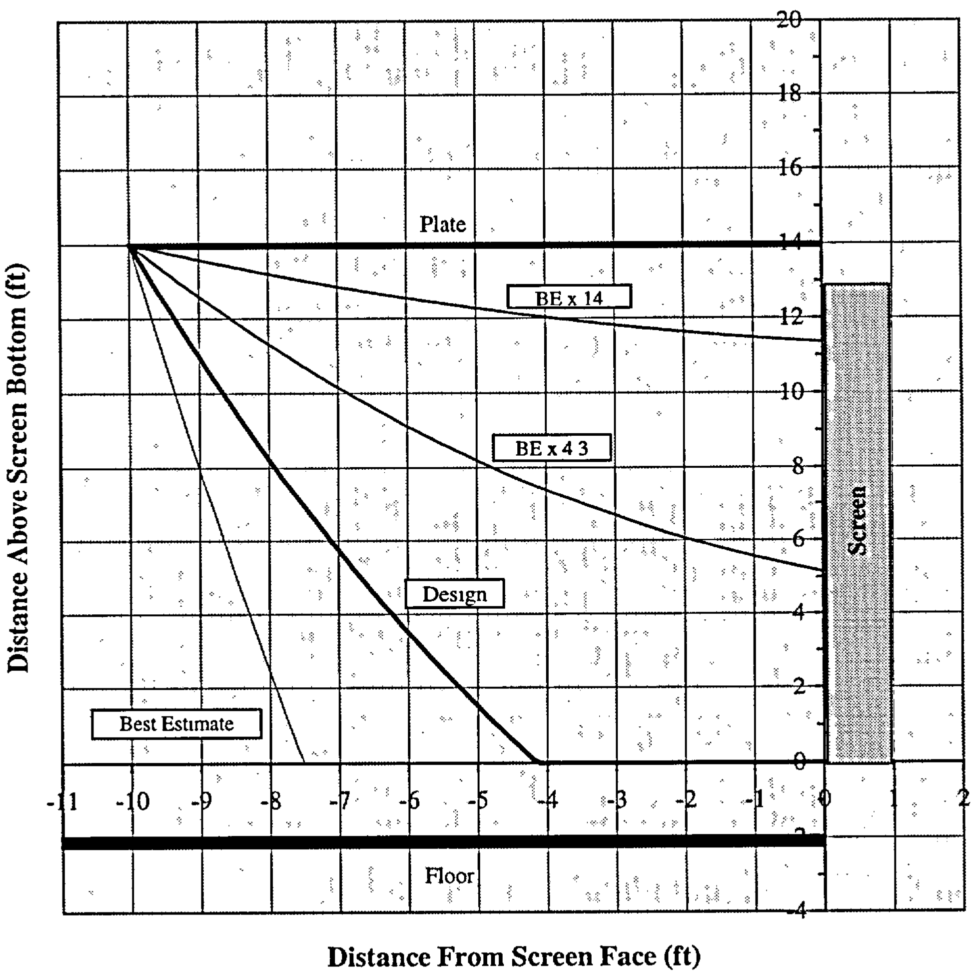


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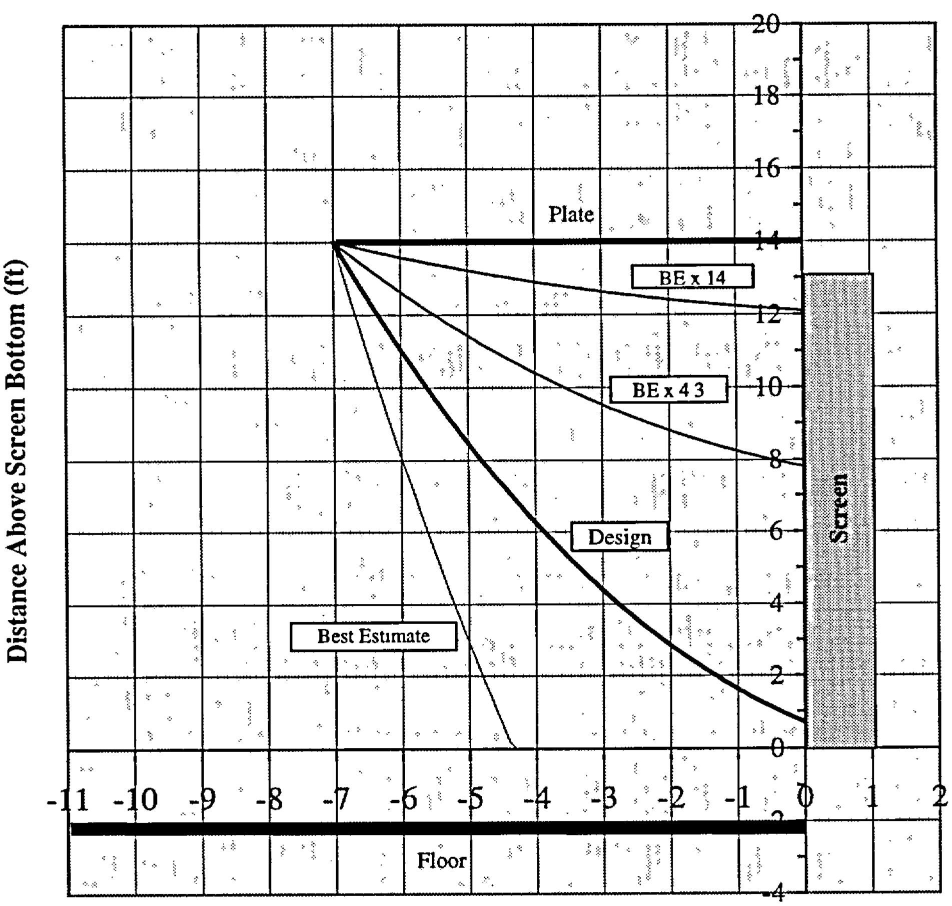
Figure 650.006-2 AP1000 Coating Debris Settling (Side)

This figure shows that the AP1000 has significant margin to accommodate uncertainty in coating settling rates. This figure shows the approach to the side of the containment recirc screen located in the corridor. Sensitivity of coating settling to debris settling rates is shown. The heavy solid line shows the AP1000 design case, which includes a margin factor of 2.00 times the reference settling data (Reference 1). Lighter lines represent sensitivity studies with greater margins. The margins applied were chosen to force the debris to settle out covering half of the screen (factor of 4.3) and covering 90% of the screen (factor of 14). Even with 90% of the screen completely blocked, either the RNS or the PXS would still be able to provide core cooling. A light line shows the settling using the reference settling data. Note that the potential screen blockage is assumed to be an average of figure 650.006-1 and -2. All of these cases assume a conservatively high flow which bounds the maximum RNS flow possible assuming both RNS pumps take suction from two screens and is unthrottled (1600 gpm/screen).



Distance From Screen Face (ft)

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Response to Request For Additional Information

RAI Number: 720.005 (Response Revision 1)

Question:

Many of the analyses described in Section A.3 performed to justify the core cooling success paths are performed using the MAAP4 code. Use of MAAP4 is based on comparisons with NOTRUMP as described in ADS4-14869, "MAAP4/NOTRUMP Benchmarking to Support Use of MAAP4 for AP600 PRA Success Criteria Analyses," for the AP600 review. Section A.2.4.2 provides a description of MAAP4/NOTRUMP benchmark to demonstrate the acceptability of MAAP4 for use in the AP600 PRA success criteria analyses. It states that the benchmarking work provides clear definition of MAAP4 capabilities and limitations.

For AP1000, the NRC staff has informed Westinghouse (letter from James Lyons to W. E. Cummins, "Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design," dated March 25, 2002) of possible deficiencies in the NOTRUMP entrainment models at the time of ADS4 actuation. Therefore the NOTRUMP-MAAP4 benchmarks may not be valid for AP1000. The NRC staff must therefore assess the validity of the entrainment models in MAAP4.

Provide justification that the MAAP4 models are appropriate for AP1000 analyses, including comparisons to appropriate experimental data for the liquid entrainment models in MAAP4 for the reactor core, upper plenum, hot legs, and ADS4. Justify that the predictions by MAAP4 for AP1000 are within the range of the test data.

A. Provide justification that the MAAP4 models are appropriate for AP1000 analyses, including comparisons to appropriate experimental data for the liquid entrainment models in MAAP4 for the reactor core, upper plenum, hot legs, and ADS4. Justify that the predictions by MAAP4 for AP1000 are within the range of the test data.

B. Describe the nucleate boiling heat transfer correlation used to model heat transfer between the passive residual heat removal (PRHR) heat exchanger (HX) tube bundle and the in-containment water storage tank (IRWST) in MAAP4. Justify that this correlation is appropriate for AP1000 by comparison to data in WCAP-12980, "AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report," Revision 3.

C. List any limitations for the application of MAAP4 to AP1000, such as the limitation described in of WCAP-14869 for the MAAP4 application to AP600.

Westinghouse Response:

A.

The MAAP4 computer code simulates the response of light water reactor systems to initiating events. It was originally developed to investigate the physical phenomena that may occur in the event of a severe accident after significant core damage. Although the emphasis in the code development has been on the severe fuel damage phase of the accident, the code has also been used to determine the thermal-hydraulic behavior prior to core damage.

MAAP4 is a fully integrated, systems accident code and includes models for important thermal-hydraulic and fission-product phenomena that may occur during a postulated accident in a pressurized water reactor (PWR) plant. The models in MAAP4 relevant to success criteria are the following:

- Reactor coolant system thermal-hydraulics
- Cladding water reaction
- Reactor core heatup
- Containment thermal-hydraulics

The version of MAAP4 used for these analyses is documented in Reference 1, which provides details of the code models, the non-AP600 benchmarking performed, and users guidance.

MAAP4 was used to determine the AP600 PRA success criteria because of its capability to analyze the reactor, passive safety-related systems, active nonsafety-related systems and the containment in an integrated fashion.

MAAP4 was benchmarked for its use for AP600, as documented in Reference 2, against the more detailed models in NOTRUMP, the Westinghouse-validated code for AP600 small-break LOCAs. A total of 19 benchmarking cases were analyzed with both MAAP4 and NOTRUMP. The first 7 cases were chosen at limiting break sizes across the spectrum of the break sizes analyzed with MAAP4. They demonstrate the basic phenomena that were identified in the PRA Phenomena Identification Ranking Tables (PIRTs). The remaining benchmarking cases were sensitivities to demonstrate the capability of MAAP4 to predict trends for different break locations, different number of core make-up tanks or accumulators, different number of automatic depressurization system lines, and different parameters affecting IRWST gravity injection.

The benchmarking work not only provides clear definitions of MAAP4 capabilities and limitations; it provides information on the response of the AP600 plant to multiple failure accidents. The response of the plant is based not only on MAAP4 calculations, but also on NOTRUMP analyses. Many of the benchmarking cases are defined based on the PRA success criteria, which means that the least required equipment is credited to show that successful core cooling is achieved.

In NUREG-1512, the NRC summarizes the MAAP4/NOTRUMP benchmarking work as follows: "The staff reviewed WCAP-14869 and evaluated Westinghouse's conclusions regarding the adequacy of MAAP4 for screening PRA sequences. The staff found that, in most cases, MAAP4 and NOTRUMP predicted similar trends for system behavior in the base cases and sensitivity analyses. On the basis of the benchmark study comparisons, the staff has determined that MAAP4 is an adequate screening tool for evaluating PRA success criteria for the AP600, subject to the limitations discussed by Westinghouse in WCAP-14869."

In the AP1000 PRA, the MAAP4 code is used as a screening tool in a manner consistent with our approach for the AP600. MAAP4 is used in the analysis of the same type of accidents where good agreement with the NOTRUMP code was found in WCAP-14869. These cases are characterized as small LOCA events with multiple failures, and include operation of either the ADS valves connected to the pressurizer, the hot legs, or both. Results typically do not show significant core uncovery, and the calculated PCTs are much less than 2200F. For cases where there is significant core uncovery such that the calculated PCTs would approach 2200F, Westinghouse performs confirmatory NOTRUMP analyses. These cases are the Thermal Hydraulic uncertainty cases that are presented in Appendix A of the PRA and confirm that the PCTs for these lower margin success sequences are less than the acceptance limit of 2200F.

The models in the MAAP4 code do not explicitly model in detail the upper plenum and hot leg entrainment discussed in this RAI. Westinghouse has submitted WCAP-15833 Revision 1 "WCOBRA/TRAC AP1000 ADS4/IRWST Phase Modeling". This report provides the Westinghouse assessment of the entrainment issue for the AP1000. Conclusions provided in this report, which are supported by detailed sensitivity studies on the importance of the upper plenum and hot leg entrainment are that these phenomena do not significantly effect the AP1000 passive safety system performance following a small break LOCA. The conservatism in the AP1000 NOTRUMP methodologies used in the thermal-hydraulic uncertainty analyses presented in Appendix A of the PRA is sufficient to account for the effects of upper plenum and hot leg entrainment.

In addition, as was shown in WCAP-15833 Revision 1, pool entrainment correlations show that liquid entrainment from the upper plenum is a strong function of the distance between the two-phase liquid level and the bottom of the hot leg. Entrainment rates fall rapidly as the liquid level approaches the top of the fuel, and for the more severe cases which exhibit core uncovery after ADS4 is actuated, entrainment is not expected to be an issue.

B. Please see the response to RAI 720.009 for a discussion of the MAAP4 model of the PRHR heat exchanger heat transfer correlation.

C. The limitations identified in Reference 2 apply to the AP1000 MAAP4 model. No additional limitations have been identified.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

References:

1. MAAP4 Modular Accident Analysis Program, User's Manual, Rev. 0, May 1994
2. MAAP4/NOTRUMP Benchmarking to Support the Use of MAAP4 for AP600 PRA Success Criteria Analysis, WCAP-14869, April 1997

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

It is stated that the models in the MAAP4 code do not explicitly model in detail the upper plenum and hot leg entrainment.

Since the staff identified possible deficiencies in the NOTRUMP entrainment models at the time of ADS4 actuation for AP1000, why is the AP600 MAAP4/NOTRUMP benchmark applicable to AP1000? Have you performed MAAP4/NOTRUMP benchmark for AP1000? If not, how do you conclude that (response to item C) the limitations identified in WCAP-14869 for AP600 apply to the AP1000 MAAP4 model and no additional limitations have been identified?

W should revise response.

Westinghouse Revised Response:

The AP600 MAAP4/NOTRUMP benchmarking cases in WCAP-14869 were selected to demonstrate the key thermal-hydraulic phenomena that occur in multiple-failure PRA accident scenarios that lead to successful core cooling. Prior to defining the benchmarking cases, PRA Phenomena Identification Ranking Tables (PIRT) were developed to define the high importance phenomena that have a controlling influence on calculating the minimum vessel water inventory. A PIRT was developed for multiple-failure scenarios with CMT (no accumulators) and a separate PIRT was developed for multiple-failure scenarios without CMT (but with accumulators). The benchmarking cases, and output parameters examined, were selected to demonstrate the phenomena that were identified in the PRA PIRT.

The AP600 MAAP4/NOTRUMP benchmarking is applicable to AP1000 because the thermal-hydraulic phenomena that control the plant response and accident progression are the same as they were for AP600. As shown in Appendix A of the AP1000 PRA, the overall plant response and specifically the core water level transients are very similar for AP1000 and AP600. Equipment modifications (e.g., larger ADS valves, larger CMT) and success criteria changes (e.g., more ADS valves are opened for crediting successful core cooling) have counter-balanced the increased power level. The phenomena that were important for AP600 continue to be important for AP1000. The PRA PIRT in Section 3 of WCAP-14869 continue to be applicable for AP1000, and thus the phenomena addressed in the AP600 benchmarking applies to AP1000 without any additional limitations.