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Your ref: Docket No. 52-006
Our ref: DCP/NRC1656

December 10, 2003

SUBJECT: Transmittal of Revised Responses to AP1000 DSER Open Items

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

Please contact me at 412-374-4728 if you have any questions concerning this submittal.

Very truly yours,

R. P. Vijuk, Manager
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/Attachments

1. List of the AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses transmitted with letter DCP/NRC1656
2. Non-Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated December 10, 2003

D063

Attachment 1

List of

Non-Proprietary Responses

Table 1 "List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1656"	
3.8.2.1-1 Revision 3 16.2-2 Revision 1 17.5-1 Revision 1 19.1.3.2-2 Revision 1 19.1.10.3-2 Revision 1 19.2.3.3-1 Revision 2 19.4-1, Revision 2	

DCP/NRC1656
Docket No. 52-006

December 10, 2003

Attachment 2

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

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.8.2.1-1 Revision 3

Original RAI Number(s): **None** (April 3, 2003, meeting summary)

Summary of Issue:

The containment vessel is an ASME metal containment. The information contained in this subsection is based on the design specification and preliminary design and analyses of the vessel. During the April 2-5, 2003 audit at Westinghouse, the applicant informed the staff that the final detailed analyses, to be documented in the ASME Design Report, are not available and will be the responsibility of the COL applicant. The staff expected that the final detailed analyses for the AP1000 steel containment would be submitted for staff review as part of the design certification process for AP1000. To complete the staff evaluation of the AP1000 steel containment design, the staff will need to audit the final detailed analyses. This is Open Item 3.8.2.1-1.

Additional NRC Comments in meeting of October 6-9, 2003

The evaluation of the containment vessel should be revised to incorporate the seismic loads described in the latest DCD. These loads were revised following the revised assumptions of shear wall stiffness (see DSER Open Item 3.7.2.3-1). Additional justification should be provided that any of the specified load combinations not evaluated are bounded by those evaluated.

The DCD should be revised to specify critical dimensions as Tier 2*. In particular, the spacing between stiffeners should be specified as Tier 2* since there is little margin in the design calculation for external pressure.

Westinghouse Response (Completely revised in Revision 2):

The detailed design calculations provided for review during the meeting on October 6-9 were initiated before the change in seismic analyses. A separate reconciliation of the new loads was prepared by Westinghouse. The revised loads have now been included in a revision to the Containment Vessel Design Specification. The detail design calculations for the containment vessel have been revised based on the updated specification. This revision also describes the selection of the load combinations and justifies why those not evaluated are less critical. These documents are available for audit.

The maximum vertical spacing of the horizontal stiffeners is added below and identified as Tier 2*.

Design Control Document (DCD) Revision:

Revise fifth paragraph of subsection 3.8.2.1.1 as follows:

The containment vessel includes the shell, hoop stiffeners and crane girder, equipment hatches, personnel airlocks, penetration assemblies, and miscellaneous appurtenances and attachments. The design for external pressure is dependent on the spacing of the hoop stiffeners and crane girder which are shown on Figure 3.8.2-1. *[The spacing between each pair of ring supports (the bottom flange of the crane girder, the hoop stiffeners, and the concrete floor at elevation 100' 0") is less than 50' 6".]**

PRA Revision:

None

NRC Follow-on comment:

Provide the frequency quantification for combined external pressure scenario and SSE events that was discussed in the October 6-9, 2003, meeting.

Westinghouse Response to NRC Follow-on comment:

Revision 3 of this response provides the requested quantification in the Attachment 3.8.2.1-1 R3-1. This information supplements the previous discussion of this topic that was provided in Attachment 1 to Westinghouse letter DCP/NRC1583 dated May 1, 2003.

Attachment 3.8.2.1-1 R3-1

AP1000 Containment Vessel Loads

Estimation of Scenario Frequency for Combination of Containment Vessel External Pressure with Safe Shutdown Earthquake

An event sequence which combines various conditions that can lead to challenge to AP1000 containment vessel structural integrity has been envisioned in the past and was analyzed to see if the containment integrity can be maintained with the postulated conditions. The end state of concern in this scenario has been the possibility of internal containment pressure dropping below the value allowed by the tech specs, combined with high wind/low temperature loads on the outside and additional occurrence of a SSE.

The objective of this paper is to estimate the frequency (expected value) of event sequences that can result in such an end state. This frequency is compared against the acceptance criterion (defined below) to see if the sequence frequency is small enough to classify it as risk-insignificant.

An event tree model is used to define and quantify the frequency of two event sequences that can potentially lead to containment vessel challenge.

Scenario 1:

A loss of AC power event occurs;
Outside temperature is -40 degrees;
One or more emergency diesel generators provide onsite AC power;
Operators fail to take actions to keep the containment pressure within technical specifications;
Containment pressure drops by 2.9 psi;
An SSE (0.3g) occurs while the containment pressure dropped by 2.9 psi;
Containment fails.

Event Tree of Figure 1 models and defines the resulting event sequence. The sequence of interest to us is sequence #6 which is postulated to lead to containment failure. The station blackout (SBO) sequence is not further pursued here.

The following frequencies/probabilities are used for calculation of the frequency of sequence #6 in Figure 1:

A - A loss of AC power event occurs;

An initiating event frequency of f1 = 7E-03/year is used. This value is taken from NUREG/CR-5750, Table 3-1

B - The outside temperature is -40 degrees;

In most states, this condition can not occur, according to actuarial weather data collected over a century. In the base case, it is postulated that this condition would occur one day in a year. Sensitivity analyses are done later to examine the impact of this assumption on the event sequence frequency.

With the current assumption, the probability q1 is calculated as 1/365 = 0.00274

C - One or more emergency diesel generators provide onsite AC power;

This event tree node is used to define the sequence more accurately (to separate it from SBO); the results are not sensitive to the value chosen (since it is almost equal to 1). A value of q2 = 0.002 is used, assuming a two-train redundant emergency diesel generator configuration. Typical failure probability for a single train would be at the order of 0.025-0.05.

D - The operators fail to take actions to keep the containment pressure within technical specifications;

Tech specs require that the containment pressure is kept within 0.2 psi of the prescribed value. Procedures, time, and equipment is available during this event sequence to maintain the containment pressure within tech specs. The performance shaping factors for this action are within normal range. Thus, the failure to perform this operator action is considered to be not likely. Consistent with NUREG/CR-1278, a human error probability (HEP) of q3 = 0.005 is use. The sensitivity of the results to this value is studied later.

E - Containment pressure drops by 2.9 psi;

With the postulated low outside temperatures, it is physically very unlikely, if not impossible (due to air cooling on the surface of the containment vessel) that the initial containment temperature will ever be 120 degrees F.; thus leading to postulated pressure drop of 2.9 psi. However, for the purposes of this study, it is assumed that this pressure drop occurs with a probability of q4 = 1.

A WGOTHIC calculation was performed to determine the containment pressure response with the containment initial temperature at as high a value as possible, and with the environment temperature as low as possible. A previous analysis was performed assuming an environment temperature of -40F and a containment atmosphere temperature of 120F, 100% relative humidity. For an operating reactor, these conditions cannot physically exist. A subsequent analysis was performed to determine the highest containment atmosphere temperature that could occur while the reactor is operating and the environment temperature is -40F.

The AP1000 WGOTHIC containment model was used with nominal heat transfer coefficients assumed between the containment atmosphere and the heat sink structures

inside containment and the containment shell. This assumption differs from the evaluation model which is used to determine the peak containment pressure following a large pipe break in containment where the heat transfer coefficients are reduced to minimize heat removal. In addition, the reactor operating heat load is modeled by a heater component in the containment. This heat load was determined by multiplying the maximum heat removal capability of the fan coolers by 120%. Thus, the resulting steady-state temperature that will be used as the initial condition for the transient calculation will be conservatively high. The environment temperature for this calculation is assumed to be -40F.

Figure 3 shows the results of this WGOTHIC analysis. The simulation was run as a transient for 50,000 seconds until a steady state was achieved. The resulting containment atmosphere temperature is approximately 75F.

To determine the minimum pressure, the following assumptions are made:

1. Initial containment conditions from steady-state analysis; 75F, 100% relative humidity

2. Internal heat sinks inside containment are assumed to be 75F.

3 Fan coolers remove operating reactor heat so that no net heat load to containment is assumed.

4. Environment temperature assumed to be -40F.

5. Heat transfer coefficients to heat sinks and containment shell are nominal.

Without an internal heat load, the containment atmosphere will cool and the pressure will decrease. The pressure response curve is shown in Figure 4. This curve shows that the pressure falls from 14.5 psia to 13.6 psia (1.1 psid) at 3600 seconds after the heat input to the containment atmosphere is terminated. This is sufficient time for operator action to prevent further pressure reduction, as discussed in AP1000 DCD Section 6.2.1.1.4. Thus the design value of 2.9 psid external pressure is very conservative.

F - An SSE (0.3g) occurs while the containment pressure dropped by 2.9 psi;

The expect value of the frequency of an SSE or higher g seismic event (0.3g or more) during a year is at the order of E-05 - E-04 for plant sites east of the Rocky mountains (NUREG-1488). For the purposes of this study a value of 0.0001/year is used. Most plant sites have an expected value lower than this.

The time of exposure to the SSE while the plant has the above conditions is taken as 8 hours (see sensitivity analyses for longer durations). It is expected that the containment pressure will be brought up to tech spec limits within a shift, after which even if an SSE occurs, the load will not fail the containment.

Thus the probability of having an SSE in an 8-hour coincidence time is calculated as q5 = 1E-04 * 8 / 8760 = 9.13E-08.

G - Containment fails;

If the above conditions hold, it is assumed that the containment will fail. Thus q6 = 1.

With the above values, the sequence frequency is calculated to 8.74E-15/year.

The acceptance criterion for this sequence is taken as 1.0E-07/year or less. This criterion is consistent with the LERF acceptance criterion in RG 1.174. Note that there is no LERF in this sequence; thus the acceptance criterion is conservative.

With the calculated sequence frequency, the sequence comfortably meets the acceptance criteria being risk-insignificant and need not be formally analyzed.

Sensitivity Analyses

In this section, the sensitivity of the sequence frequency to three important assumptions in the base model is analyzed. These are:

The one cold day per year assumption in q2;
The human error failure probability value in q3;
The 8-hour SSE coincidence time in q5.

If 30 cold days per year is assumed (q2 = 0.0822), the sequence frequency becomes 2.62E-13/year;

If an error factor of 10 is assumed for the operator action (q3 = 0.05), the sequence frequency becomes 8.74E-014/year;

If the SSE coincidence time is taken as 24 hours following the initiating event (q5 = 2.74E-08), then the sequence frequency becomes 2.62E-04/year.

Each of these frequencies still meets the acceptance criteria comfortably.

There is so much margin in the calculated sequence frequency that even if all the conservatisms in the above sensitivities are piled up, the frequency increases by a factor of almost three orders of magnitude (30 * 10 * 3 = 900), the sequence frequency becomes 7.87E-12/year, which still meets the acceptance criteria by a large margin. In this case the margin factor is 12700 (1E-07/7.87E-12), which is very large.

A second scenario may also have been considered. It is discussed in the next section.

Second Scenario

The following scenario can be envisioned:

Scenario 2:

A SSE event occurs;
AC power event occurs due to SSE;
Outside temperature is -40 degrees;
One or more emergency diesel generators provide onsite AC power;
Operators fail to take actions to keep the containment pressure within technical specifications;
Containment pressure drops by 2.9 psi;
An aftershock event with at least 0.3g SSE (0.3g) occurs while the containment pressure dropped by 2.9 psi;
Containment fails.

Event Tree of Figure 2 models and defines the resulting event sequence. The sequence of interest to us is sequence #7 which is postulated to lead to containment failure. The station blackout (SBO) sequence is not further pursued here.

The initiating event frequency is taken as f2 = 1E-04/year, as discussed in Scenario 1. The initiating event causes loss of offsite power (due to failure of ceramic insulators whose seismic fragilities are lower). Thus, q7= 1.

The probability of an aftershock of magnitude 0.3g or higher is very difficult to estimate. Generally, the aftershocks are lower in magnitude than the initiating earthquake. For the purposes of this calculation, two values are used:

q8 = 0.01

And

q8 = 0.5.

The scenario in Figure 2 is quantified by using the first value. The resulting scenario frequency is 1.37E-11/year. This is well below the acceptance criterion.

When the second value is used, the scenario frequency becomes 6.84E-10/year. This value is also below the acceptance criteria.

The sensitivity cases with 30 days of cold days, and 10 times higher operator action probability applied individually to the base case still meet the acceptance criteria with the first probability for q8. If the second probability is used (q8=0.5), then these sensitivity cases still meet the acceptance criterion.

Figure 1 Event Tree Model and Calculation of the Frequency of Scenario 1 (page 1 of 2)

Loss of AC	Cold Day	Onsite AC	Operators Fail	Pressure Drops	SSE Occurs	Containment Fails	Seq. No.	End State	Frequency
A	B	C	D	E	F	G			
							1	OK	
							2	OK	
f1 0.007							3	OK	
		0.998					4	OK	
			q3 0.005				5	OK	
	q1 2.74E-03			q4 1			6	CONT-FAILS	8.74E-15
					q5 9.13E-08	q6 1	7	SBO	
		q2 0.002							

SBO = Station blackout (loss of offsite and onsite emergency AC power).

Figure 1 (continued)

Description of event tree nodes

A	A loss of AC power event occurs;
B	The outside temperature is -40 degrees;
C	One or more emergency diesel generators provide onsite AC power;
D	The operators fail to take actions to keep the containment pressure within technical specifications;
E	The containment pressure drops by 2.9 psi;
F	An SSE (0.3g) occurs while the containment pressure dropped by 2.9 psi;
G	The containment fails.

Values used to quantify event sequence #6

f1	0.007	NUREG/CR-5750, Table 3-1
q1	0.002739726	One day per year
q2	0.002	Estimate - does not affect results
q3	0.005	Estimate
q4	1	Given as occurred
q5	9.13242E-08	
q6	1	Given as occurred

Figure 2 Event Tree Model and Calculation of the Frequency of Scenario 2 (page 1 of 2)

SSE Occurs	Loss of Offsite Power	Cold Day	Onsite AC	Operators Fail	Pressure Drops	Aftershock SSE Occurs	Containment Fails	Seq. No.	End State	Frequency
A	B	C	D	E	F	G	H			

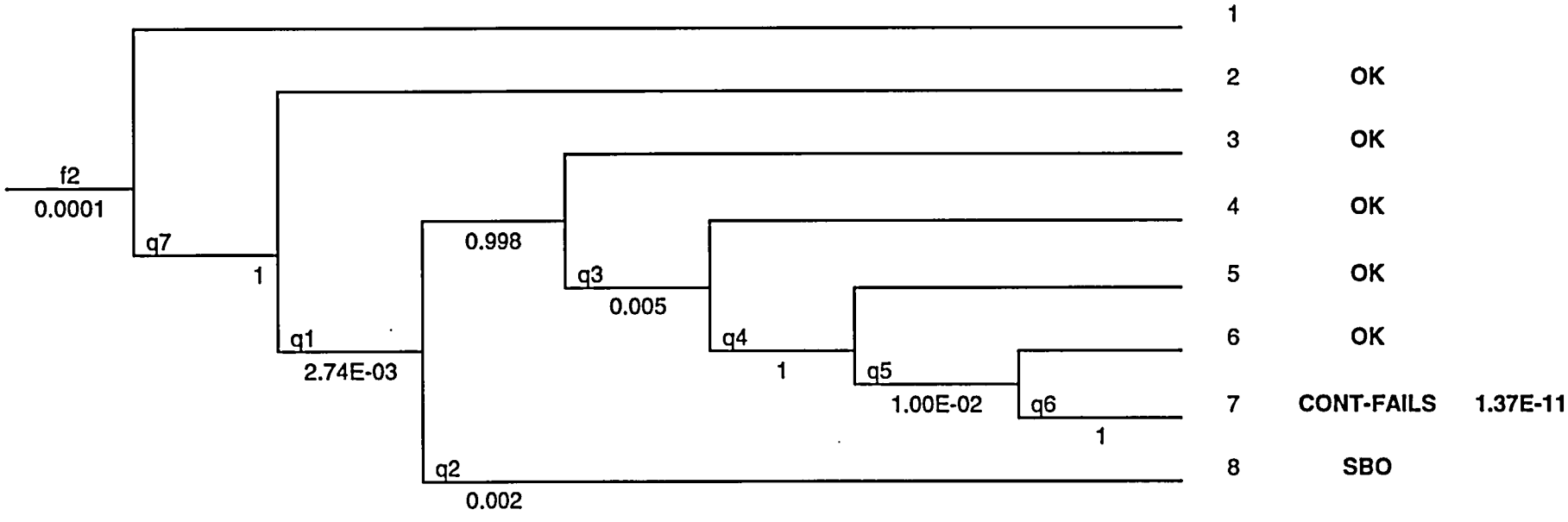

Figure 2 (continued)

Description of Event Tree Nodes

A	A SSE event occurs;
B	AC power event occurs due to SSE;
C	Outside temperature is -40 degrees;
D	One or more emergency diesel generators provide onsite AC power;
E	Operators fail to take actions to keep the containment pressure within technical specifications;
F	Containment pressure drops by 2.9 psi;
G	An aftershock event with at least 0.3g SSE (0.3g) occurs while the containment pressure dropped by 2.9 psi;
H	Containment fails.

Values used to quantify event sequence #7

f2	1.00E-04	same as Scenario 1
q1	0.00274	One day per year
q2	0.002	Estimate - does not affect results
q3	0.005	Estimate
q4	1	Given as occurred
q8	0.01	estimated
q6	1	Given as occurred
q7	1	Given as occurred

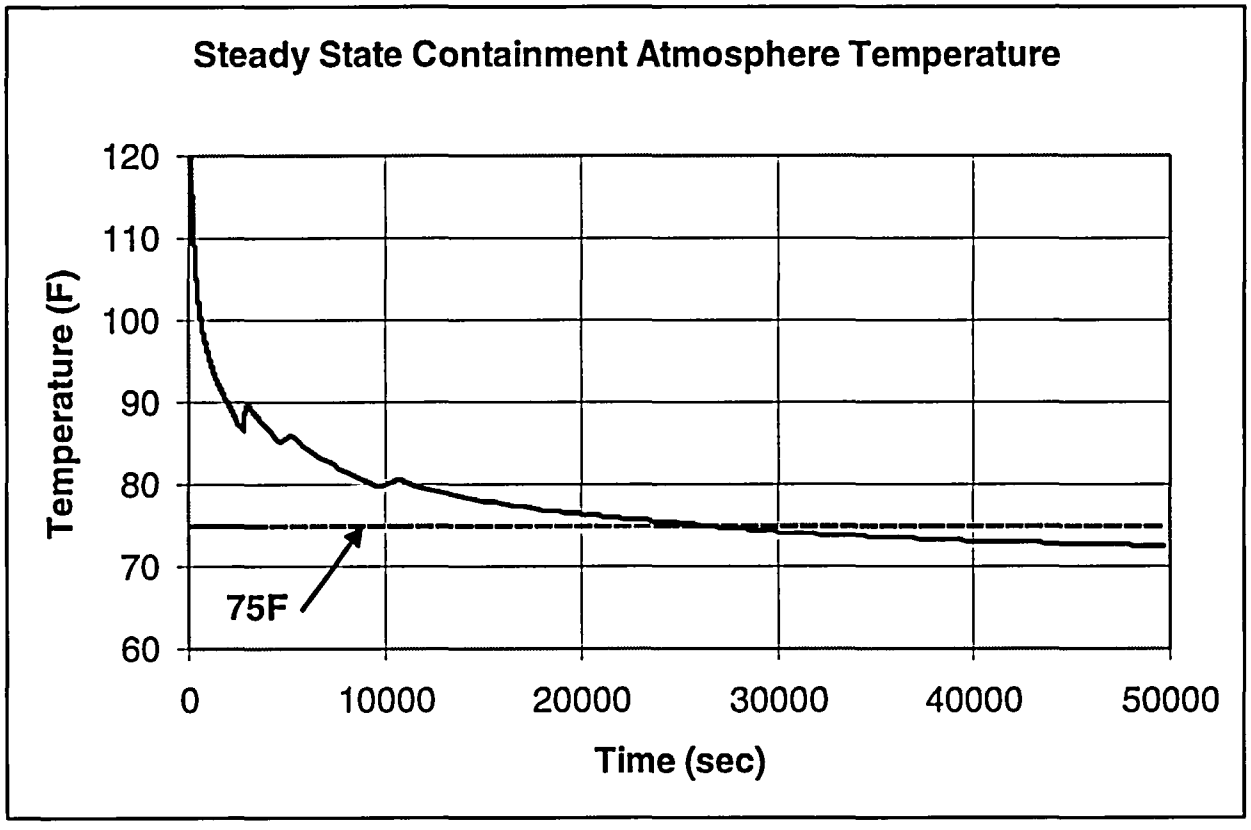

Figure 3: Steady-State Operating Temperature for Containment Atmosphere

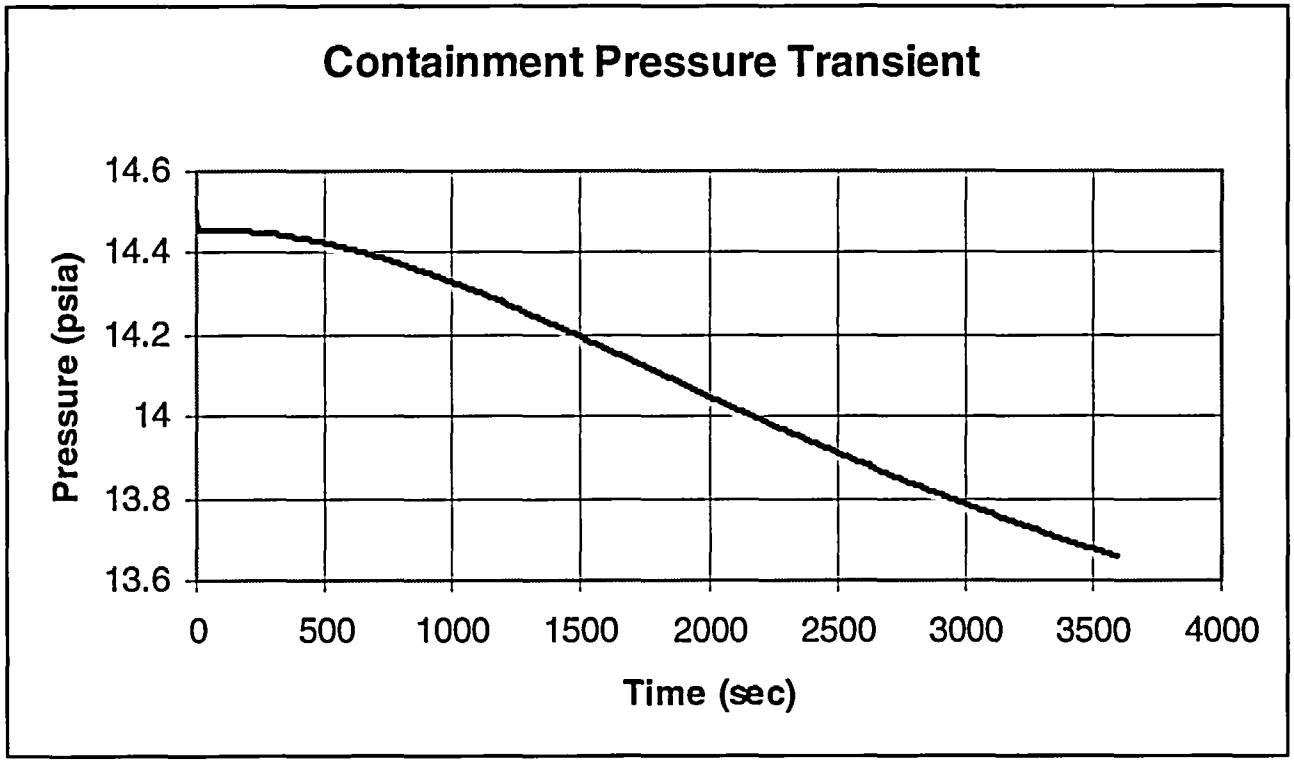

Figure 4: Containment Minimum Pressure Transient

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 16.2-2 Response Revision 1

Original RAI Number(s): None

Summary of Issue:

The TS action requirements for the CMT, PRHR, and IRWST PXS subsystems allow 72 hours for loss of a redundancy, which is consistent with STS 3.5.2; however, the Bases for the PXS LCOs seem to indicate that only one subsystem at a time is affected. The AP1000 TS do not identify what the appropriate actions are in the event the plant does not meet two or more PXS specifications (e.g., 3.5.1, 3.5.2, 3.5.4 and 3.5.6) concurrently. The Bases for the PXS LCOs also seem to indicate that DBA assumptions regarding ECCS functions may not be met in such cases. Pending clarification of the Bases, the staff's review of the PXS TS action requirements is considered incomplete. This is Open Item 16.2-2.

Westinghouse Response:

The approach for the response to this Open Item is to first provide a comparison of the AP1000 PXS Technical Specifications (TSs) and the STS ECCS TSs for current plants to demonstrate the consistent approach in following the STS model and philosophy to develop the PXS TSs. After comparing the two sets of TSs, the next step in responding to this Open Item is to identify the allowable PXS equipment Conditions in the AP1000 TSs and to confirm an acceptable PXS operational capability, consistent with the current STS, during the most limiting combinations of allowable Conditions for the PXS equipment. The table developed for this second step shows that appropriate actions are specified in the PXS TSs when LCOs for more than one PXS TS are not met, even for the most limiting design basis accident, and that like the STS, conditional TS actions are not required.

Questions related to understanding the PXS operational capabilities while in multiple TS Action statements may result from two possible sources, a potentially confusing sentence in the Bases LCO discussion for two PXS components and the structure of Required Actions in the AP1000 PXS TSs (due to PXS simplification) that do not require treatment or evaluation of the PXS equipment on a specifically identified train basis. These two aspects will be addressed as part of this response.

The Technical Specification Bases for TSs 3.5.1 and 3.5.2 will be revised to change one sentence in the Bases LCO discussion to clarify when design basis accident assumptions regarding emergency core cooling system functions are met.

The Bases discussions for these PXS LCOs were intended to be equivalent to and consistent with the STS Bases discussions in NUREG-1431, Rev. 2. The Bases Background discussions for these two TS each discuss design basis mitigation functions, consistent with the STS Bases. The AP1000 Bases also attempted to improve the Bases Background discussion completeness by also including a few sentences on PRA mitigation performance for beyond-design-basis equipment failures. The wording mentioned in the original LCO discussion for the accumulator

and CMT Bases was trying to jointly characterize subsystem performance assumptions for both design basis and beyond-design-basis cases in one summary statement. However, the statement appears to add confusion when trying to understand specific design basis assumptions for these two LCOs, as indicated in the discussion for the Open Item. Therefore, the two revisions shown below will be made to the Bases LCO discussion for AP1000 TSs 3.5.1 and 3.5.2 to eliminate the confusing wording related to the interaction between the various PXS subsystems, and to make them more consistent with the STS LCO Bases.

As a result of the evaluation in part two to respond to this Open Item, the Condition A statement for TS 3.5.6 and the associated Bases discussion, which currently allows a loss of actuation redundancy in one of the four containment recirculation valve flow paths, will be revised slightly to allow a loss of actuation redundancy for either one of the four recirculation flow paths OR one of the four IRWST injection line flow paths.

The original Condition was determined to be overly restrictive considering credible redundant actuation valve malfunctions that could occur, and was identified as part of the systematic review of allowable Conditions in part two of this response. This is equivalent to a loss of actuation redundancy in one train in the ECCS and is also consistent with the loss of redundancy allowed in other PXS components such as the redundant, parallel CMT discharge isolation valves or PRHR discharge isolation valves. The 72-hour Completion Time for the original IRWST Condition statement still applies to the revised Condition statement.

Based on discussions with the NRC reviewer in understanding the issue for this Open Item, the evaluation presented to respond to this issue focuses on the various loss-of-coolant accidents (LOCAs) requiring safety injection and core cooling. Other plant events such as rod ejection, reactor vessel failure, loss of secondary coolant, and steam generator tube rupture also require a similar safety injection mitigation function and have been considered, but they are bounded by the limiting event for the purposes of this response evaluation.

Decay heat removal for the mitigation of non-LOCA events is provided by the PRHR (AP1000 TS 3.5.4), while the other PXS components perform safety injection and core cooling functions required to mitigate LOCAs. PRHR operation is functionally equivalent to the decay heat removal provided by Auxiliary Feedwater (AFW) in the STS 3.7.5 for current plants. The failure to meet other PXS LCOs is relatively independent of the PRHR status since the design basis mitigation functions for the other PXS equipment are for LOCA events. Therefore, PRHR is included in the comparison of TSs and in the list of allowable Conditions for completeness of both tables, but does not need to be addressed in the Open Item response evaluation.

The other PXS equipment - accumulators (TS 3.5.1), Core Makeup Tanks (CMTs) (TS 3.5.2), and IRWST (TS 3.5.6) – each provide different design basis safety injection and core cooling mitigation functions and the operation of these other PXS components is less complex than the ECCS equipment in current plants. This simplification and the resulting structure of the AP1000 TSs eliminates the need for plant operators to perform any AP1000 PXS equipment train operability evaluations, which are required in STS 3.5.2 for the ECCS train operability determination in current plants, as discussed later. The evaluation in step two will help to clarify the PXS operational capability in the event that more than one LCO is not satisfied.

AP1000 TS Comparison to the STS

The AP1000 PXS TSs were developed using the STS ECCS TSs as models. The corresponding TSs for AP1000 and STS are summarized in Table 1. The purpose of comparing the AP1000 TSs and the STS is to help confirm the equivalence and consistency between the two documents.

The relationship between the AP1000 PXS TSs is similar to the relationship between the STS ECCS TSs in that the PXS and ECCS equipment in the various Section 3.5 TSs do not provide functional redundancy to each other for design basis accidents. The individual TSs for both AP1000 and the STS are written to preclude the need for conditional Required Actions, where the operability of ECCS equipment in one TS would depend on the operability of components in another TS, for circumstances when two or more different PXS or ECCS LCOs are not met simultaneously. The AP1000 TSs are written similarly to and consistent with the STS, although the AP1000 design provides greater PXS simplification, component safety injection, and core cooling functional independence compared to current plants in the STS.

Support system operability requirements for both the AP1000 PXS equipment and the STS ECCS equipment are addressed separately in Section 3.5 PXS and ECCS TSs. The Required Actions in the STS and AP1000 TS are consistent with the requirements in LCO 3.0.6 for support systems and in TS 5.5.15 (STS) / TS 5.5.8 (AP1000) for the Safety Function Determination Program. The AP1000 provides greatly reduced dependencies on support systems such as ac electrical power and compressed air, requiring only the availability of dc electrical power for component actuation (ADS MOVs and ADS/IRWST/containment recirculation squib valves) and for monitoring instrumentation. The other PXS components (CMTs and PRHR) actuate by fail-open valves or by natural processes that open check valves (accumulators, IRWST injection, and containment recirculation).

Current plants in the STS are more limiting than the AP1000 in terms of support system interrelations between the ECCS equipment in Section 3.5 of the STS. The RWST in STS 3.5.4 provides the water inventory for the ECCS trains in STS 3.5.2, although there are no conditional Required Actions needed in STS 3.5 even with this support relationship. This equivalent water inventory support relationship does NOT exist between the AP1000 PXS TSs, so there is greater independence between the PXS component TSs than for current plants in the STS.

As shown in Table 1, the AP1000 PXS design includes the accumulators, CMTs, and IRWST (for safety injection functions), and PRHR (for non-LOCA decay heat removal). Therefore, to be exactly consistent with the STS format for safety injection equipment, individual TSs are provided for the accumulators and IRWST, as shown.

The accumulators for both AP1000 and STS perform equivalent functions, so the TSs are almost identical for this intermediate-pressure safety injection source. The AP1000 TS has the same train identification approach for the accumulators as the STS, with Conditions for one accumulator and for more than one accumulators (trains) inoperable. Therefore train operability

for component is easily identifiable in the TS and identical to the STS since there is one tank in each train, which is similar to current plants.

The IRWST provides low-pressure safety injection and includes injection lines and containment recirculation lines. Therefore, the Conditions and Required Actions related to tank operability (boron, temperature, volume) are almost identical to STS 3.5.4. However, the AP1000 IRWST TS also includes two additional, and relatively simple Conditions and Required Actions associated with the injection line and recirculation line actuation valves. The AP1000 TS includes Conditions and Required Actions that allow one of four redundant containment recirculation valve paths (one of two paths in one of two trains) and one of two redundant injection paths (trains) to be inoperable. Therefore, train operability for this component is easily identifiable since there is one common tank, each injection line and containment recirculation line is one train, and each train has redundant, parallel actuation valve paths.

Since the remaining PXS safety injection components, the CMTs, also required a TS, AP1000 TS 3.5.2 was written to be consistent with the STS methodology, and to replace STS 3.5.2. STS 3.5.2 is far the more complex since it includes the multiple ECCS trains in current plants, and requires evaluating the operability of the ECCS high-head, low-head, and possibly intermediate head safety injection (SI) pumps, along with the associated heat exchanger and numerous isolation valves in each train. The AP1000 TS 3.5.2 is relatively simple since it only includes tank operability Conditions (boron and temperature), piping high point voiding Condition, and redundant discharge isolation/actuation valve Condition and associated Required Actions for each Condition.

The simplicity of the AP1000 TS 3.5.2 eliminates the need for the operator to perform the more complex ECCS train operability evaluation of STS 3.5.2. For example, Condition C requires the operator to determine if "100% of the ECCS flow equivalent to a single OPERABLE ECCS train..." This determination involves extensive evaluations of available components in the two ECCS trains and the associated judgements about which SI functions are provided by which redundant trains, including the numerous valves, heat exchangers, support system operability such as cooling water, ac electrical power, and dc electrical power.

Train-specific Conditions, Required Actions, and train operability evaluations are inherent, but much less obvious in the various AP1000 PXS TSs, due to the simplicity of the PXS design. But the PXS operability requirements and the resulting Conditions and Required Actions in the event that the various LCOs are not met are consistent with the STS.

Therefore, train operability for this component is easily identifiable in the TS. For example, each CMT and associated inlet and outlet piping is one train, and each train has redundant, parallel discharge actuation valve flow paths. However, specifically evaluating CMT train operability is not required by the operators since the TS implicitly and directly addresses train operability without the specific need for an operator evaluation.

The AP1000 Automatic Depressurization System (ADS) is also included in this evaluation. The ADS is physically part of the Reactor Coolant System (RCS) and the TS is appropriately located in Section 3.4 of the AP1000 TS. However, the ADS TS is included in Table 1 since the ADS

valves have a design basis safety injection and core cooling function following loss of coolant accidents (LOCAs). Therefore, a discussion of allowable ADS Conditions is also included in part two of the response to this Open Item.

DSER OI 16.2-2, Table 1

Equivalent AP1000 and STS Technical Specifications

AP1000	STS
3.5.1 Accumulators	3.5.1 Accumulators
3.5.2 CMTs, Operating 3.5.3 CMTs, Shutdown	3.5.2 ECCS (injection pump trains), Operating 3.5.3 ECCS, Shutdown
3.5.6 IRWST, Operating 3.5.7 IRWST, Shutdown, Mode 5 3.5.8 IRWST, Shutdown, Mode 6	3.5.4 RWST
3.4.12 ADS, Operating 3.4.13 ADS, Shutdown, RCS Intact 3.4.14 ADS, Shutdown, RCS Open	3.4.11 Pressurizer Power-Operated Relief Valves
3.5.4 PRHR, Operating 3.5.5 PRHR, Shutdown	3.7.5 AFW

Allowable AP1000 TS Conditions

The second part of this response involves evaluating the limiting combinations of the various PXS equipment Conditions allowed by each TS in the event that the individual LCOs are not met, and confirming the acceptability of the limiting combinations of plant Conditions. Table 2 lists the allowable TS Conditions that do not require entry into LCO 3.0.3 or plant shutdown for the various PXS and ADS TSs considered (including PRHR). Table 2 identifies two bounding combinations of allowable Conditions, one for the shortest Completion Time and one for the longest Completion Time, and also lists the remaining Conditions that were considered, but not included in the two limiting cases.

Case 1 lists the most limiting set of allowable PXS equipment Conditions with an 8-hour Completion Time. Case 2 lists the most limiting set of allowable Conditions with a 72-hour Completion Time. Case 3 lists all remaining allowable Conditions that were not included in the two limiting cases. Each Condition with the 8-hour and 72-hour Completion Times includes a brief summary of the equipment status for the Condition and an associated note that characterizes the expected status of the PXS component in that Condition. For example, the

discussion may describe the component as failed, having degraded injection performance, or having degraded actuation redundancy. The combinations of Conditions for the two bounding evaluation cases are summarized in the Evaluation discussion. The 1-hour IRWST Condition is also included in the 8-hour case since only two IRWST Conditions exist and they both fit best in the 8-hour Completion Time case.

In selecting a combination of Conditions for each case, the more restrictive component Condition in terms of component performance for the specific Completion Time that is allowed by TSs is included. For some components, two Conditions may be listed for a specific case for simplification, as discussed. The remaining, less restrictive Conditions for each PXS component are listed in Case 3, which allows all Conditions to be displayed in the table for completeness. This is helpful in showing that the most restrictive Condition was used in Cases 1 and 2. One allowable Condition with an intermediate Completion Time for the CMTs is not included in the evaluation since the Condition is bounded by other Conditions that have more limiting Completion Times for the CMTs. For the cases that show only one set of train failures, it is always assumed that the first failure is in Train A. The mirror image degradation or loss of components can also occur, but is not shown for simplicity since the effects are the same.

In evaluating PXS operability when multiple LCOs are not met, all categories of LOCA events were considered, as well as other plant events that would require safety injection. The limiting LOCA event used for the evaluation of the allowable PXS Conditions is the direct vessel injection (DVI) line break. This limiting line break disables one complete train of PXS equipment - the accumulator, CMT, IRWST injection line, and containment recirculation line that all share the same DVI flow path. This results in only one train of PXS equipment available for injection through the other intact DVI line. Therefore, the limiting combination of equipment for the two Completion Times cases are evaluated for the DVI line break event.

DSER Open Item 16.2-2, Table 2

PXS Component and Allowable TS Conditions	Completion Time	Case 1 8-Hour Completion Time		Case 2 72-Hour Completion Time		Case 3 Other Allowable Conditions	
		Train A	Train B	Train A	Train B	Train A	Train B
Accumulators – both operable		Degraded or failed performance	OK	Degraded boration	OK	OK	OK
- Boron OOS	72 hrs			(1)			
- Other than boron	8 hrs	(2)					
CMT – both operable		Failed or degraded performance	OK	Degraded redundancy	OK	Degraded performance	Degraded performance
- Outlet isol valve	72 hrs			(3)			
- Temp / boron OOS	72 hrs					(4)	
- 2 temp / boron OOS	8 hrs					(4a)	(4a)
- High point gases	24 hrs					Not bounding	Not bounding
- Inoperable for other reasons	8 hrs	(5)					
IRWST Inj - 2 paths		Degraded performance or redundancy	OK	OK	OK	OK	OK
- Boron / temp / >97%	8 hrs	(6)					
- Injection MOV	1 hr	(7)					
Recirc – 2 paths		OK (8)	OK	Degraded redundancy	OK	OK	OK
- Recirc MOV	72 hrs			(9)			
ADS – 10 paths		OK (8)	OK	Degraded redundancy	OK	OK	OK
- 1 path inop	72 hrs			(10)			
- 1 and either 2/3 inop	72 hrs			(11)			
PRHR		Not evaluated	Not evaluated	Not evaluated	Not evaluated	Not evaluated	Not evaluated
- Outlet isol valve	72 hrs					Degraded redundancy (12)	OK
- Gutter isol valve	72 hrs					Degraded redundancy (13)	OK
- High point gases	24 hrs					Not bounding	Not bounding
- Other	8 hrs					Fail or degraded performance (14)	OK

Notes for Table 2

(1) Degraded accumulator RCS boration, but insignificant impact on injection when boron is out of specification low. Unlikely for boron to be out of specification high. Any potential boron deviations are expected to be slight, considering that pressure and water volume are verified daily.

(2) Degraded or failed accumulator injection performance. Degraded performance would most likely be due to slight deviations in water volume or gas pressure due to leakage. Any potential deviations are expected to be slight, considering that pressure and water volume are verified daily. Failure could occur due to discharge MOV misalignment that could fail or significantly degrade the injection capability. While injection performance may be impaired or the accumulator may be inoperable, this condition is only allowed for a very short time interval. One accumulator is sufficient for any break except a cold leg LOCA and leak-before-break incorporation significantly reduces the likelihood of an RCS loop break. PRA shows success with one accumulator for a large LOCA caused by spurious ADS actuation and that no accumulators are required for a small LOCA, assuming that one CMT is available.

(3) Degraded CMT actuation redundancy, which does not impact CMT injection flow. One of the two parallel outlet isolation valves for the CMT is inoperable, but the CMT is still capable of functioning, assuming no single failure occurs. For this case, the 72-hour Completion Time is based on the small likelihood of an event occurring, combined with the likelihood that a single failure will occur upon actuation, which is consistent with a loss of ECCS redundancy in the STS.

(4) Degraded CMT injection performance. Increase in CMT temperature results in a slight reduction in the injection mass flow rate. A reduction in boron concentration reduces the shutdown boration capability, but does not impact injection flow. In either case, it is likely that more than the required amount of boron and injection flow will be available to meet the conditions assumed in the safety analyses. For this case, the 72-hour Completion Time is acceptable based on the small likelihood of an event occurring, combined with the relatively small expected impact on the injection or boration capability. Since the degraded redundancy was considered more limiting for the 72-hour case, this Condition was included with other allowable Conditions.

(4a) Degraded CMT injection performance for both CMTs. This is the same condition as in Note 4, except that it applies to both CMTs. For this condition, both CMTs are expected to inject, with a slight reduction in the injection mass flow rate or slightly degraded boration. This condition is less limiting than Note 5, so it was included with other allowable Conditions.

(5) CMT injection is inoperable for some reason other than boron concentration or water temperature. This could potentially prevent injection, but some postulated causes such as the inlet test isolation valve being inadvertently closed, are expected to be able to be quickly corrected. A potential cause of an MOV problem is the valve being inadvertently manually closed for some reason such as being left closed after discharge valve inservice testing. The

inlet MOV is expected to be relatively easy to restore to the open position and the valve has a confirmatory open signal on a CMT actuation, and it is expected to be operable to open since it was just recently closed. In the event of inoperability due to failure of monitoring instrumentation, the CMT is expected be capable of performing its injection function, but the surveillance requirements cannot be performed. The PRA shows successful core cooling with only accumulator and IRWST injection for small LOCAs. The likelihood of an event with the CMT inoperable for such a short Completion Time is relatively small.

(6) Degraded IRWST injection performance. For these parameters, the injection performance is only very slightly degraded in the most credible postulated condition. The water volume may be slightly below the 100% level, but above 97%, which has a very slight reduction in injection due to the small decrease in injection elevation head and an insignificant impact on total PXS injection volume. Boron deviations have a slight impact on boration shutdown capability, but have no impact on injection performance. Boron deviations are not expected to be significant since it is extremely difficult to have a large boron change in such a large tank. Water temperature deviations have only a very slight impact on injection performance, due to reduced gravity injection head. IRWST volume and temperature also impact the heat sink capability for the PRHR, but this in not a significant impact since the potential parameter variations are not expected to be large. The relatively short Completion Time for these parameter deviations prevents these conditions from existing for a long period of time, since the parameters are expected to be able to be easily restored to operable condition within this short time frame.

(7) Degraded actuation redundancy for IRWST injection. One of the two redundant IRWST injection lines may not be operable since the common IRWST injection line isolation MOV is not fully open. A possible cause is the MOV being inadvertently manually closed for some reason, and the valve is expected to be relatively easy to restore to the open position. In addition, the valve has a confirmatory open signal on a safety injection. Although a closed valve fails one of the two IRWST injection lines, the redundant IRWST injection line is fully operable, which is relatively unaffected for all events except a DVI line break on the side of the operable IRWST injection line. In addition, the associated containment recirculation lines can provide injection via reverse flow in the affected line, back into the IRWST, through the IRWST, and back out the unaffected IRWST injection line into the RCS. The short Completion Time is provided in recognition of the impact of a DVI line break on the side of the operable IRWST line, and also based on the expected time to restore the injection line MOV to a fully-open position.

(8) The only Conditions with shorter Completion Times than 72 hours have Required Actions that require plant shutdown or entry into LCO 3.0.3, and are not included in this table, as discussed in the evaluation.

(9) Degraded actuation redundancy for containment recirculation. One of the four containment recirculation paths may not be operable since an isolation MOV is not fully open. Three of the four containment recirculation paths are still operable, so recirculation is still capable of functioning even with a single failure, except for one limiting event which is a DVI line break in the opposite IRWST injection flow path. A possible cause is the MOV being inadvertently manually closed for some reason, and the valve is expected to be relatively easy to restore to the open position. In addition, the valve has a confirmatory open signal on a low IRWST level.

For this case, the 72-hour Completion Time is based on the small likelihood of an event occurring, combined with the likelihood that a single failure will occur upon actuation, which is consistent with a loss of ECCS redundancy in the STS.

(10) Degraded ADS actuation redundancy. One of the 10 paths of ADS is inoperable, but the ADS can still perform its design basis function, assuming no single failures. The limiting ADS failure is an inoperable Stage 4 path. If other paths are inoperable, the impact on ADS performance is significantly less, as seen in Item (11) that allows one Stage 1 and either a Stage 2 or Stage 3 path to be inoperable. An ADS Stage 4 flow path can also be inoperable because an isolation MOV is not fully open. A possible cause of an MOV problem is the valve being inadvertently manually closed for some reason. The MOV is expected to be relatively easy to restore to the open position and the valve has a confirmatory open signal on a ADS Stage 4 actuation, and it is expected to be operable to open since it was most likely just recently closed. For small break LOCAs the limiting single failure is the loss of one Stage 4 flow path. The PRA shows that adequate core cooling can be provided with the failure of up to seven flow paths (all ADS Stage 1 to 3 and one ADS Stage 4). The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS Stage 4 valves to open. All of the ADS Stage 1, 2, 3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation. For this condition with a single failed ADS path, the 72-hour Completion Time is based on the small likelihood of an event occurring, combined with the likelihood that a single failure will occur upon actuation, which is consistent with a loss of ECCS redundancy in the STS.

(11) Degraded ADS actuation redundancy. For this case where a Stage 1 valve flow path and either a Stage 2 or 3 valve flow path are simultaneously inoperable, the ADS can still perform its design basis function, assuming no single failures. In this case, ADS performance still meets the design basis, assuming that no single failure occurs. As mentioned in Item (10), the performance in this case is bounded by the single failure of a Stage 4 valve allowed in Item (10), and this Required Action provides additional plant operational flexibility in the event of multiple equipment malfunctions. This demonstrates the increased flexibility allowed by the AP1000 PXS design. For this condition with a single failed ADS path, the 72-hour Completion Time is based on the small likelihood of an event occurring, combined with the likelihood that a single failure will occur upon actuation, which is consistent with a loss of ECCS redundancy in the STS.

PRHR Notes - Presented only for completeness and NOT included in evaluation of PXS design basis safety injection for the LOCA events.

(12) Degraded PRHR actuation redundancy, which does not impact PRHR decay heat removal for non-LOCA events. One of the two parallel outlet isolation valves for the PRHR is inoperable, but the PRHR is still capable of functioning, assuming no single failure occurs. For this case, the 72-hour Completion Time is based on the small likelihood of an event occurring, combined with the likelihood that a single failure will occur upon actuation, which is consistent with a loss of ECCS redundancy in the STS.

(13) Degraded gutter isolation valve redundancy occurs for return of condensate to the IRWST following an event with steaming into containment. One of the two, series gutter drain isolation valves (to the containment sump) is inoperable, but the remaining isolation valve can still function to isolate the drain path to the sump so that condensate is returned to the IRWST, assuming no single failure occurs. For this case, the 72-hour Completion Time is based on the small likelihood of an event occurring, combined with the likelihood that a single failure will occur upon actuation, which is consistent with a loss of ECCS redundancy in the STS.

(14) The PRHR HX is inoperable for some reason other than the discharge isolation valves. This could potentially prevent PRHR decay heat removal. Some postulated causes such as the inlet test isolation valve being inadvertently closed are expected to be able to be quickly corrected. The potential cause of an MOV problem is the valve being inadvertently manually closed for some reason such as being left closed after discharge valve inservice testing. The inlet MOV is expected to be relatively easy to restore to the open position and the valve has a confirmatory open signal on a PRHR actuation, and it is expected to be operable to open since it was just recently closed. The PRA shows that the PRHR HX is not required assuming that passive feed and bleed is available. Passive feed and bleed for beyond-design-basis events in the PRA uses the ADS for bleed and the CMTs/accumulators/ IRWST for feed. The effectiveness of feed and bleed cooling has been demonstrated in analysis and evaluations performed to justify PRA success criteria. The 8 hour Completion Time is based on the availability of passive feed and bleed cooling to provide RCS heat removal. The likelihood of an event with the PRHR inoperable for such a short Completion Time is relatively small.

Evaluation Summary

Case 1 represents the allowable TS Conditions for the AP1000 where, like the STS, design basis protection may not be available for a short time period without requiring an immediate plant shutdown. For these Conditions, it is credible to restore some of the more likely postulated component malfunctions within the Completion Time, as discussed in the notes for Table 2. While the equipment inoperability disables the component function, the short Completion Time results in a small impact on plant risk. The risk of remaining in a stable plant condition and allowing this short time period to restore the ECCS or PXS equipment to operable status has been judged to be acceptable. In addition, the very short Completion Time for this case also makes it extremely unlikely for multiple PXS components to become simultaneously inoperable.

The trade-off in overall plant safety in this situation is the likelihood of an event occurring during the short Completion Time while in relatively stable, steady-state plant conditions with inoperable PXS or ECCS components, compared to the impact on plant safety due to the potential increased likelihood of an event while conducting the sequence of evolutions and plant equipment changes required for a plant shutdown transient.

A similar approach is allowed by the STS. For example, one or more ECCS trains can be inoperable for 72 hours, provided that the equivalent flow of one ECCS train can be confirmed to be available. If an emergency diesel-generator simultaneously becomes inoperable, the plant is allowed to continue to operate for a short period of time (4 hours) before declaring the

affected ECCS components in the affected diesel-generator electrical buses inoperable and requiring a plant shutdown. This is a reasonable time to evaluate the Conditions and attempt to restore the likely causes of the inoperable equipment before initiating a shutdown transient that also impacts plant risk.

The same approach has been followed in developing the AP1000 TSs and allowing the Conditions identified in Case 1 to exist with the relatively short Completion Time before a plant shutdown must be performed if the equipment is not restored to operable status.

Case 2 represents the allowable TS Conditions for the AP1000 where there is loss of design basis redundancy, and is consistent with the 72-hour Completion Time allowed in the STS, as stated in the first sentence of the Open Item discussion. For these Conditions, the design basis can be met assuming that no single failures occur. Therefore, the allowable PXS Conditions are consistent with the allowable STS Conditions for this case.

Case 3 consists of miscellaneous allowable Conditions in the AP1000 that are simply listed for completeness, but have not been included in Case 1 or Case 2 since they are not the limiting allowable Conditions for either case. The two Conditions indicated are not discussed in the notes since they are bounded by the evaluated Conditions.

Therefore, the TS Required Actions when more than one core cooling TS Limiting Conditions for Operation (LCO) is not met are equivalent for both the AP1000 PXS TSs and the STS ECCS TSs. For this reason, there is no need for Required Actions in the AP1000 TSs that are conditional upon the operability of the other PXS components, consistent with the Required Actions for the STS ECCS equipment.

In addition, while both AP1000 PXS and STS ECCS subsystems provide design basis mitigation functions, as well as mitigation for beyond-design-basis accidents, the passive AP1000 PXS design provides greater defense-in-depth through this redundant functionality for beyond-design-basis accident functions than current plants. The AP1000 PXS design utilizes a much simpler subsystem design for each PXS component, so that with significantly fewer safety-related components to malfunction, the probability that two or more of the AP1000 PXS TS LCOs will not be satisfied simultaneously is much lower than for current plants.

Design Control Document (DCD) Revision:

DCD Chapter 16, Basis 3.5.1, page B 3.5-3

LCO

This LCO establishes the minimum conditions necessary to ensure that sufficient accumulator flow will be available to meet the necessary acceptance criteria established for core cooling by 10 CFR 50.46 (Ref. 5). These conditions are:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

DCD Chapter 16, Basis 3.5.2, page B 3.5-9

LCO

This LCO establishes the minimum conditions necessary to ensure that sufficient CMT flow will be availableto meet the initial conditions assumed in the safety analyses. The volume of each CMT represents 100% of the total injected flow assumed in LOCA analysis. If the injection line from a single CMT to the vessel breaks, no single active failure on the other CMT will prevent the injection of borated water into the vessel. Thus the assumptions of the LOCA analysis will be satisfied.
For non-LOCA analysis, two CMTs are assumed. Note that for non-LOCA analysis, the accident cannot disable a CMT.

DCD Chapter 16, TS 3.5.6, page 3.5.6-1

3.5.6 In-containment Refueling Water Storage Tank (IRWST) – Operating

LCO 3.5.6 The IRWST, with two injection flow paths and two containment recirculation flow paths, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

	CONDITION		REQUIRED ACTION	COMPLETION TIME
A.	One IRWST injection line actuation valve flow path inoperable. OR One containment recirculation line actuation valve flow path inoperable.	A.1	Restore the inoperable actuation valve flow path to OPERABLE status.	72 hours

DCD Chapter 16, TS 3.5.6, page B 3.5.6-1, Background, paragraphs 2 and 3

The IRWST has two injection flow paths. The injection paths are connected to the reactor vessel through two direct vessel injection lines which are also used by the accumulators and the core makeup tanks. Each path includes an injection flow path and a containment recirculation flow path. Each injection path includes a normally open motor operated isolation valve and two parallel actuation lines each isolated by one check valve and one squib valve in series.

The IRWST has two containment recirculation flow paths. Each containment recirculation path contains two parallel actuation flow paths, one path is isolated by a normally open motor operated valve in series with a squib valve and one path is isolated by a check valve in series with a squib valve.

DCD Chapter 16, TS 3.5.6, page B 3.5.6-3, Action A.1

If an IRWST injection line actuation valve flow path or a containment recirculation line actuation valve flow path is inoperable, then the valve actuation flow path must be restored to OPERABLE status within 72 hours. In this condition, three other IRWST injection or containment sump recirculation flow paths are available and can provide 100% of the required flow assuming a break in the direct vessel injection line associated with the other injection train, but with no single failure of the actuation valve flow path in the same injection or sump recirculation flow path. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

PRA Revision:

None

NRC Additional Comments:

Based on subsequent discussions with the TS reviewer following submittal of the original response to this open item, two initial clarifications were requested by the reviewer:

- Clarification of the word "outlet" when referring to the IRWST isolation valve
- Clarification of the plant response to actions if SR 3.5.6.5 confirms that power is not removed to and IRWST isolation MOV

In a follow-up telephone call on 10/28/03 to discuss the Westinghouse response, there was further discussion on one aspect of the response related to identified backup capabilities between the accumulator and CMT during small LOCA events as described in the Background discussions for AP1000 TSs 3.5.1 and 3.5.2. The reviewer had concerns about the TS impact of simultaneous unavailability of both an accumulator and a CMT. The reviewer believed that some method was needed to make a decision on the potential need for a conditional response in the event that multiple PXS Conditions are simultaneously entered for the accumulators and CMTs. The AP1000 design and implementation of the PXS TSs is technically different from the specific ECCS train determination required in TS 3.5.2 of the STS. So there may be technical justification for implementing a conditional response for these two AP1000 PXS TSs.

Westinghouse mentioned that some PRA analysis of the PXS components had previously been done to support the response to another open item that may be helpful in evaluating this condition, and that this PRA work would be reviewed to determine the relevance of addressing this specific question.

Westinghouse Response to NRC Additional Comments:

The Bases discussion for SR 3.5.6.4 and 3.5.6.5 will be changed to delete the word "outlet" since all other references in Section 3.5 of the AP1000 TSs identify these valves as "motor operated IRWST isolation valves" and this word is not used anywhere else.

Condition C of TS 3.5.6, 3.5.7, and 3.5.8 and the associated discussion of the Actions in the Bases will be revised to also make the 1 hour Completion Time applicable if it is determined that power is not removed from the motor operated IRWST isolation valve. This approach for power not removed is consistent with the approach and treatment for similar power operated valves in Section 3.5 of the AP1000 TSs and the STS. The Completion Time is reasonable considering the consequences of Condition C, where these valves are inoperable due either to mis-positioning or power not being removed. In both cases, it is also expected to be possible to restore the valve to OPERABLE status within the specified Completion Time.

In reviewing TSs 3.5.6, 3.5.7, and 3.5.8 as part of this response, an editorial error was found in the IRWST volume in SRs 3.5.6.2 and 3.5.8.2. The correct value for IRWST minimum volume is 73,900 ft3 instead of the current value of 78,900 ft3. This corrected value is now consistent with the value in Table 6.3-2 (Sheet 2 of 2).

Westinghouse looked at the existing PRA analysis work for a different open item which was applicable to this issue and concluded that additional analysis would be required to justify the exclusion of conditional responses for simultaneous multiple PXS Condition entry, although it is expected that PRA analysis could justify this approach.

Considering the very low probability of simultaneous entry into CMT and accumulator TS Conditions, it was judged more appropriate to simply incorporate conditional Completion Times for three specific CMT and accumulator Conditions that would require them to address the interactions as described in the TS Bases for these two TSs.

In evaluating which accumulator and CMT Conditions to consider, the backup capability only exists between the accumulator and CMTs for a small LOCA event. Therefore, a conditional Completion Time is only needed for Condition B of TS 3.5.1 for the accumulators and for Conditions C and E of TS 3.5.2 for the CMTs. The other Conditions do not require conditional Completion Times. For example, minor tank parameter deviations (like temperature or boron) do not have any significant impact on small LOCA backup capability for the component. And for the CMTs, voiding at the tank high point has no impact on a small LOCA CMT performance where it provides a backup for the accumulator since significant RCS voiding occurs as a consequence of the event.

For these three Conditions, the conditional Completion Times were incorporated with a 1 hour time limit if simultaneous Conditions were entered, and the previously existing Completion Time when simultaneous Conditions are not entered, as described in each TS. The 1 hour conditional Completion Time is reasonable, as discussed in the Bases justification for each TS.

Design Control Document (DCD) Revision:

See the attached revisions to the following TSs and Bases:

- Bases for SRs 3.5.6.4 and 3.5.6.5
- Condition C of TS 3.5.6, 3.5.7, and 3.5.8 and the associated Bases for Action C.1 for each TS
- SRs 3.5.6.2 and 3.5.8.2 (No changes are required to the SR 3.5.7.2 since it references SR 3.5.6.2.)
- Condition B of TS 3.5.1 and the associated Bases for Action B.1
- Conditions C and E of TS 3.5.2 and the associated Bases for Actions C.1 and E.1

PRA Revision:

None.

REVISION TO BASES for SRs 3.5.6.4 and 3.5.6.5

SR 3.5.6.4

This surveillance requires verification that each motor operated isolation valve is fully open. This surveillance may be performed with available remote position indication instrumentation. The 12 hour Frequency is acceptable, considering the redundant remote indication and alarms and that power is removed from the valve operator.

SR 3.5.6.5

Verification is required to confirm that power is removed from each motor operated IRWST isolation valve each 31 days. Removal of power from these valves reduces the likelihood that the valves will be inadvertently closed. The 31 day Frequency is acceptable considering frequent surveillance of valve position and that the valve has a confirmatory open signal.

REVISION TO TS 3.5.6

Actions

C.	One motor operated IRWST isolation valve not fully open. OR Power is not removed from one or more motor operated IRWST isolation valves.	C.1	Restore motor operated IRWST isolation valve to fully open condition with power removed from both valves.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.6.1	Verify the IRWST water temperature is < 120°F.	24 hours
SR 3.5.6.2	Verify the IRWST borated water volume is > [73,900] cu. ft.	24 hours

REVISION TO TS 3.5.7

Actions

C.	Required motor operated IRWST isolation valve not fully open. OR Power is not removed from required motor operated IRWST isolation valve.	C.1	Restore required motor operated IRWST isolation valve to fully open condition with power removed.	1 hour

REVISION TO TS 3.5.8

Actions

C.	Required motor operated IRWST isolation valve not fully open. OR Power is not removed from required motor operated IRWST isolation valve.	C.1	Restore required motor operated IRWST isolation valve to fully open condition with power removed.	1 hour

Surveillances

SR 3.5.8.2	Verify the IRWST and refueling cavity water total borated water volume is > [73,900] cu. ft.

REVISION TO BASES for ACTIONS C.1 of TSs 3.5.6, 3.5.7, and 3.5.8

BASES

ACTIONS

C.1

If the motor operated IRWST isolation valves are not fully open or valve power is not removed, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open with valve power removed in 1 hour. This Completion Time is acceptable based on risk considerations.

REVISIONS TO TS 3.5.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration outside limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	8 hours if Condition C or E of LCO 3.5.2 has not been entered OR 1 hour if Condition C or E of LCO 3.5.2 has been entered
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce RCS pressure to ≤ 1000 psig.	6 hours 12 hours
D. Two accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

REVISION TO TS 3.5.2

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One CMT inoperable due to one CMT outlet isolation valve inoperable.	A.1	Restore outlet isolation valve to OPERABLE status.	72 hours
B.	One CMT inoperable due to one or more parameters (water temperature, boron concentration) not within limits.	B.1	Restore water temperature or boron concentration to within limits.	72 hours
C.	Two CMTs inoperable due to water temperature or boron concentration not within limits.	C.1	Restore water temperature or boron concentration to within limits for one CMT.	8 hours if Condition B of LCO 3.5.1 has not been entered OR 1 hour if Condition B of LCO 3.5.1 has been entered
D.	One CMT inoperable due to presence of non-condensible gases in one high point vent.	D.1	Vent noncondensible gases.	24 hours
E.	One CMT inoperable for reasons other than Condition A, B, C, or D.	E.1	Restore CMT to OPERABLE status.	8 hours if Condition B of LCO 3.5.1 has not been entered OR 1 hour if Condition B of LCO 3.5.1 has been entered

REVISIONS TO BASES 3.5.1 ACTIONS

Action B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 8 hours. With one accumulator inoperable, the remaining accumulator is capable of providing the required safety function, except for one low probability event (large cold leg LOCA) discussed in the background section. The effectiveness of one accumulator is demonstrated in analysis performed to justify PRA success criteria (Ref. 4). The analysis contained in this reference shows that for a range of other events including small LOCAs and large hot leg LOCAs that with one accumulator unavailable the core is adequately cooled. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk (Ref. 7).

The 8 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time is reasonable since the CMTs are required to be available to provide small break LOCA mitigation (i.e., entry into Condition C or E of LCO 3.5.2 has not occurred). The effectiveness of backup CMT injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA, the injection from one CMT without any accumulator injection supports adequate core cooling. This analysis provides a high confidence that with the unavailability of one accumulator, the core can be cooled following design bases accidents.

The 1 hour Completion Time, in the case with simultaneous entry into Condition C or E of LCO 3.5.2, requires very prompt actions to restore either the accumulator or the CMT to OPERABLE status. This Completion Time is considered reasonable because of the low probability of simultaneously entering these multiple PXS Conditions and the very small likelihood of a LOCA occurring at the same time.

REVISIONS FOR BASES 3.5.2 ACTIONS

Action C.1

With two CMTs inoperable due to water temperature or boron concentration, at least one CMT must be restored to within limits in 8 hours. The deviations in these parameters are expected to be slight, considering the frequent surveillances and control room monitors. A Completion Time of 8 hours is considered reasonable since the CMTs are expected to be capable of performing their safety function with slight deviations in these parameters and the accumulators are required to be available for LOCA mitigation (i.e., entry into Condition B of LCO 3.5.1 has not occurred). The effectiveness of accumulator injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA, the injection from one accumulator without any CMT injection supports adequate core cooling. This analysis provides a high confidence that with the unavailability of two CMTs due to water temperature or boron concentration deviations, the core can be cooled following design bases accidents.

The 1 hour Completion Time, in the case with simultaneous entry into Condition B of LCO 3.5.1, requires very prompt actions to restore either the CMT or the accumulator to OPERABLE status. This Completion Time is considered reasonable because of the low probability of simultaneously entering these multiple PXS Conditions and the very small likelihood of a LOCA occurring at the same time.

Action E.1

With one CMT inoperable for reasons other than Condition A, B, C, D, operation of the CMT may not be available. Action must be taken to restore the inoperable CMT to OPERABLE status within 8 hours. The remaining CMT is sufficient for DBAs except for LOCA in the OPERABLE CMTs DVI line. The 8 hour Completion Time is based on the required availability of injection from the accumulators (provided that entry into Condition B of LCO 3.5.1 has not occurred) to provide SI injection. The effectiveness of accumulator injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA, the injection from one accumulator without any CMT supports adequate core cooling. This analysis provides a high confidence that with the unavailability of one CMT, the core can be cooled following design bases accidents.

The 1 hour Completion Time, in the case with simultaneous entry into Condition B of LCO 3.5.1, requires very prompt actions to restore either the CMT or the accumulator to OPERABLE status. This Completion Time is considered reasonable because of the low probability of simultaneously entering these multiple PXS Conditions and the very small likelihood of a LOCA occurring at the same time.

REVISIONS FOR BASES FOR SRs 3.5.6.4 and 3.5.6.5

SR 3.5.6.4

This surveillance requires verification that each motor operated isolation valve is fully open. This surveillance may be performed with available remote position indication instrumentation. The 12 hour Frequency is acceptable, considering the redundant remote indication and alarms and that power is removed from the valve operator.

SR 3.5.6.5

Verification is required to confirm that power is removed from each motor operated IRWST isolation valve each 31 days. Removal of power from these valves reduces the likelihood that the valves will be inadvertently closed. The 31 day Frequency is acceptable considering frequent surveillance of valve position and that the valve has a confirmatory open signal.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 17.5-1 Revision 1

Original RAI Number(s): None

Summary of Issue:

In an effort to ensure that the COL action items in DCD 17.5, associated with D-RAP and O-RAP, are accomplished in a manner consistent with the guidance contained in SECY 95-132, the applicant should provide a COL action item to reflect conformance with the SECY 95-132 guidance. This is DSER Open Item 17.5-1.

Westinghouse Response:

SECY-95-132, item F, addresses the reliability assurance program. It specifies that with respect to the O-RAP, that failures of safety related SSCs related to maintenance will be dealt with by the maintenance rule. Failures that are related to design or operational errors will be dealt with by the QA requirements of 10 CFR Part 50. We have added a statement to the DCD section 17.5 to reflect the guidance on design and operational errors as requested by the staff.

Revision 1 to this response adds reference to SECY-95-132 in DCD section 17.5 as indicated below.

Design Control Document (DCD) Revision:

17.5 Combined License Information Items

The Combined License applicant will address its design phase Quality Assurance program, as well as its Quality Assurance program for procurement, fabrication, installation, construction and testing of structures, systems and components in the facility. The quality assurance program will include provisions for seismic Category II structures, systems, and components.

The COL applicant will establish PRA importance measures, the expert panel process, and other deterministic methods to determine the site-specific list of SSCs under the scope of RAP.

Combined License applicant is responsible for integrating the objectives of the O-RAP into the Quality Assurance Program developed to implement 10 CFR 50, Appendix B. This program will address failures of safety related, risk-significant SSCs that result from design and operational errors in accordance with SECY-95-132 Item F.

The Combined License applicant will address its Quality Assurance program for operations.

The following activities are represented in Figure 17.4-1 as "Plant Maintenance Program."

The Combined License applicant is responsible for performing the tasks necessary to maintain the reliability of risk-significant SSCs. Reference 8 contains examples of cost-effective maintenance enhancements, such as condition monitoring and shifting time-directed maintenance to condition-directed maintenance.

The Maintenance Rule (10 CFR 50.65) is relevant to the Combined License applicant's maintenance activities in that it prescribes SSC performance-related goals during plant operation.

In addition to performing the specific tasks necessary to maintain SSC reliability at its required level, the O-RAP activities include:

- Reliability data base – Historical data available on equipment performance. The compilation and reduction of this data provides the plant with source of component reliability information.
- Surveillance and testing – In addition to maintaining the performance of the components necessary for plant operation, surveillance and testing provides a high degree of reliability for the safety-related SSCs.
- Maintenance plan – This plan describes the nature and frequency of maintenance activities to be performed on plant equipment. The plan includes the selected SSCs identified in the D-RAP.

PRA Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.1.3.2-2 Revision 1

Original RAI Number(s): None

Summary of Issue:

Important Insights from Level 2 PRA and Supporting Sensitivity Analyses

An additional PCS-related failure mode is plugging of the drains near the floor of the annulus around the containment shell. Drain plugging can lead to accumulation of PCS water in the annulus, eventually reaching the baffle plate in the annulus and interrupting the air circulation. The availability of the PCS annulus drains will be confirmed every two years in accordance with the TSs. In the AP600 PRA, PCS failure was dominated by blockage of the PCS annulus drain lines, which was estimated to have a probability of 1E-04. This failure mechanism is not modeled in the AP1000 PRA, but at that same failure probability would have a corresponding containment failure frequency of about 2E-11/yr. Inclusion of this failure mode would substantially increase the frequency of CFLs in the AP1000. However, the frequency of CFL would remain less than 0.1 percent of the total containment failure frequency. Although not a key failure mode, for completeness of the PRA model, the staff believes that Westinghouse should include this failure mechanism within the AP1000 PRA. This is Open Item 19.1.3.2-2.

Westinghouse Response:

Attachment 43E titled Effect of Containment Air-Cooling Failure on Plant Risk is included in the Chapter 43 of the AP1000 PRA to account for the effect of this failure mechanism on plant LRF.

Design Control Document (DCD) Revision:

None

PRA Revision:

Section 43.7.4 is added in Chapter 43.

Attachment 43E is added in Chapter 43, Release Frequency Quantification.

43.7.4 Treatment of PCS Annulus Drain Plugging in the AP1000 PRA

The AP600 PRA included failure of the PCS drains as a failure mode for the containment. This was a conservative assumption in the PRA and represented the only long-term containment failure mode in the AP600 PRA.

In the AP1000, the PCS drain failure mode was dropped as a containment failure mode since it is really not expected to result in containment failure, and other potential long-term containment failures related to PCS water failure are present in the PRA. By definition of the drain failure case, PCS water flow over the containment shell is guaranteed. Cooling the containment shell with water, even without the air flow through the annulus, is expected to remove sufficient heat from the shell to prevent containment failure. Higher water film temperature, and therefore higher containment pressure, is expected if there is no air flow. The containment pressure may approach, and even exceed the design basis pressure. However, given that multiple drain failures are needed to produce this accident sequence, it is considered to be beyond the design basis. As such, the pressure is not expected to challenge the containment ultimate pressure, or even Service Level C.

Therefore, dropping PCS drain blockage as a containment failure mode for the AP1000 is justifiable, given that it is an overly conservative containment failure mode.

ATTACHMENT 43E

Effect of Containment Air-Cooling Failure on Plant Risk

This attachment discusses the effect of containment air-cooling failure on plant risk given the success criteria that air-cooling alone is sufficient to prevent containment failure for accidents studied in the base AP1000 PRA model.

When PCS is modeled by fault trees to be used in the at-power CDF event trees (under the event tree top event CHR), to identify and collect the late containment failure (LCF) end states for sequences, it includes only water cooling function. This function serves both as short –term and long term (24-72 hours) cooling. The objective of introducing LCF end state was to collect those success sequences where only air cooling by PCS is deemed to be sufficient to avoid core damage, and both the water cooling by PCS and normal RHR are unavailable. This collection is stored under the LCF end-state with a frequency of 6.92E-08/year, which is not a CDF end state, but represents the uncertainty in the sufficiency of containment cooling solely by PCS air-cooling.

Failure of air-cooling is deemed to be less likely than the mechanical and actuation failure modes already accounted for in the PCS water cooling fault tree models. Thus, this failure mode is not assigned a failure probability. Moreover, other supplies of water are expected to be available from the fire protection system, demineralized water system, ancillary water system and temporary sources (fire trucks or water buffaloes) that can be brought on line by the operators to avoid dependence on air only cooling.

In the context of AP1000 PRA Chapter 6, the following success criteria is in effect for containment cooling:

Containment cooling either by

1. "Water cooling mode" of PCS

   or

2. Decay heat removal mode of normal RHR

   or

3. "Air cooling mode" of PCS

is sufficient to prevent core damage during the mission time specified for CDF event trees. Moreover, the probability of failure of all three of these functions for an other wise "success" sequence is deemed to be very small. Thus, this containment cooling function is not queried in the CDF event trees for CDF purposes.

If these LCF sequences were to lead to core damage, then the same sequences would also lead to a late LRF consequence. The frequency of additional late LRF (it is also CDF) introduced by failure of air cooling on top of failure of water cooling and normal RHR cooling (for otherwise success end states) is estimated below for different values of air cooling reliability.

The table below shows the relation between assuming different values for air-cooling failure probability, and the resulting increase in plant CDF/LRF:

Air cooling Failure Probability	Current LCF with air cooling success	Increase in LRF (also CDF) if LCF and failure of air cooling occurs	comparative increase in base LRF	Risk Significance
0.0001	6.92E-08	6.92E-12	very small	Insignificant
0.001	6.92E-08	6.92E-11	very small	Insignificant
0.01	6.92E-08	6.92E-10	3.5%	Insignificant

From this table, one sees that with any reasonable value for the air-cooling failure probability, the increase in LRF is not risk significant, increase of CDF being even less significant.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.1.10.3-2 Revision 1

Original RAI Number(s): None

Summary of Issue:

Major Contributors to System Failures

The major causes of reactor cavity flooding failure and hydrogen igniter failure in AP1000 have not been provided. Such information is useful for identifying major contributors to system failure and confirming that reasonable measures have been taken to reduce risk. The staff will request that the applicant provide this information for AP1000. This is Open Item 19.1.10.3-2.

Westinghouse Response:

The major causes of reactor cavity flooding and hydrogen igniter failure are given in Tables 1 and 2. These are the top cutsets that fail these systems.

For reactor cavity flooding failure, almost 90% of the failure probability come from common cause failure of recirculation MOVs to open or operator action failure to open recirculation MOVs.

For hydrogen igniter failure, 75% of the failure probability comes from the first four contributors in Table 2, which are common cause failure of hydrogen igniters, failure of 12 VAC distribution panel, failure of manual actuation, and common cause failure of sensors.

NRC Follow-on Comment:

The dominant contributors to reactor cavity flooding failure in AP600 (common cause failure of strainers and common cause failure of actuation software) contribute substantially less in AP1000.

Westinghouse Response:

Response to NRC follow-on comment:

To respond to this comment, we examined the cutsets in Table 1. As a result of this examination, the following conclusions are reached:

1. The reactor cavity flooding event tree node in the containment event tree is modeled by the event tree named IWF. This event tree was based on similar event trees that modeled sump recirculation after IRWST is emptied in events such as LOCAs. The fault tree logic was changed to represent the cavity flooding that can only be performed by an

operator action, and using only at least one of two sump recirculation lines that contain MOVs. The other two sump recirculation lines have check valves that open only in the wrong direction, and thus can not be used for cavity reflooding.

2. In AP1000, two important design changes are implemented. These changes are:

   i) the MOVs are normally open (they needed to be opened in AP600). Thus, their failure to open should not appear in the IWF cutsets.

   ii) two pairs of squib valves in the recirculation paths are of different types (high pressure versus low pressure valves). Thus their CCF component groups are different. In AP600, all squib valves belonged to the same CCF component group.

These design changes affect the cutsets of IWF fault tree, as discussed below.

3. The MOV failure to open failure mode has been removed from the AP1000 IWR fault tree model. Thus, no MOV failure to open cutsets should appear. This is a major change from the AP600 PRA.

   Although the MOV failure mode is removed at the component level, the CCF failure of MOVs, which is modeled at a higher level in the fault tree is inadvertently left in the fault tree. This resulted in the first cutset in Table 1. This cutset is no longer applicable to the IWF system, and should be discarded. When this cutset is discarded, the IWF system failure probability is lowered by a factor of 2. Thus, this oversight is conservative.

4. The second cutset refers to the operator action to open recirculation valves to flood reactor cavity. The phrase describing this operator action in Table 1 is an attempt at shortening the actual description, and is misleading since it refers to MOVs. The actual longer description of this operator action, as taken from AP1000 HRA section, is "Failure to recognize the need and failure to open the recirculation valves to flood reactor cavity after core damage".

5. The third cutset refers to the CCF of two low pressure squib valves on the two lines that can provide the cavity reflooding function. An interesting observation is that the CCF probability of failure of 2/2 valves (in AP1000) is higher than CCF probability of 2 of 8 valves, which was the case for AP600 (all eight high pressure squib valves of the IRWST/recirculation) were in the same CCF component group in AP600.

6. Cutsets 4, 5, and 6 refer to CCF of strainers, and PMS associated with actuation of squib valves.

7. Cutset 7 is a modeling leftover from the original fault tree that is used to generate IWF; it refers to failure of IRWST level signal to open recirculation lines, which is only applicable to automatic actuation of recirculation when IRWST level is low. This does not apply to

cavity flooding, which is actuated only by manual operator action which appears in cutset 2.

8. The remaining cutsets show various combinations of squib valve or their support system (for actuation) failures.

9. Note that the PMS failures associated with failure to open MOVs (which show up in AP600) do not show up in the current cutsets.

10. There are some lower probability cutsets (cutsets 35-38) containing test and maintenance (T&M) unavailability of two buses. These cutsets may not be applicable since tech specs could prohibit both buses being in T&M. However, due to the small probability of these cutsets, no attempt is made to remove them.

Table 1a below shows the IWF cutsets taken from Table 1, after the above mentioned revisions are made. Although the IWF system failure probability after this revision is lower, the original failure probability is kept as is; the PRA model is not revised.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Table 1. AP1000 PRA REACTOR CAVITY FLOODING CUTSETS

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	4.40E-03	55.75	CCF OF RECIRC MOVs TO OPEN	4.40E-03	IWX-MV-GO
2	3.40E-03	43.08	OPERATOR FAILURE TO OPEN RECIRC MOVs	3.40E-03	REN-MAN03
3	5.80E-05	0.73	CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
4	1.20E-05	0.15	CCF OF STRAINERS IN IRWST TANK	1.20E-05	IWX-FL-GP
5	1.10E-05	0.14	CCF OF PMS ESF OUTPUT LOGIC SOFTWARE	1.10E-05	CCX-PMXMOD1-SW
6	8.62E-06	0.11	CCF OF EPO BOARDS IN PMS	8.62E-06	CCX-EP-SAM
7	4.78E-06	0.06	CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
			OPER. FAILS TO ACT. SUMP RECIRC GIVEN IRW LEVEL SIGNAL FAILURE	1.00E-02	REN-MAN04
8	2.13E-06	0.03	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
9	1.28E-06	0.02	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
10	1.28E-06	0.02	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
			HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
11	1.20E-06	0.02	SOFTWARE CCF OF ALL CARDS	1.20E-06	CCX-SFTW
12	7.67E-07	0.01	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
			RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
13	4.38E-07	0.01	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDS1TM
14	4.38E-07	0.01	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDD1TM
15	4.38E-07	0.01	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
16	4.38E-07	0.01	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
17	3.50E-07	0	SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
18	3.50E-07	0	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
19	2.63E-07	0	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
20	2.63E-07	0	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
21	2.63E-07	0	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDS1TM
22	2.63E-07	0	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDD1TM
23	2.50E-07	0	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
			FAILURE OF OUTPUT DRIVER	1.71E-04	IRCEP118ASA

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
24	2.50E-07	0	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			FAILURE OF THE POWER INTERFACE BOARD (###EP####SA)	1.71E-04	IRDEP118BSA
25	2.10E-07	0	SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
26	2.10E-07	0	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
27	2.06E-07	0	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
			CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PMBMOD1
28	2.06E-07	0	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PMAMOD1
29	1.50E-07	0	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
			FAILURE OF OUTPUT DRIVER	1.71E-04	IRCEP118ASA
30	1.50E-07	0	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
			FAILURE OF THE POWER INTERFACE BOARD (###EP####SA)	1.71E-04	IRDEP118BSA
31	1.41E-07	0	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
			CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PMB030
32	1.41E-07	0	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PMA030
33	1.24E-07	0	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
			CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PMBMOD1

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
34	1.24E-07	0	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
			CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PMAMOD1
35	9.00E-08	0	BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDS1TM
36	9.00E-08	0	BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDD1TM
37	9.00E-08	0	BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDS1TM
38	9.00E-08	0	BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDD1TM
39	8.49E-08	0	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
			CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PMB030
40	8.49E-08	0	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
			CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PMA030

Table 2. AP1000 PRA HYDROGEN IGNITOR CUTSETS

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	3.20E-04	27.71	CCF OF THE HYDROGEN IGNITERS	3.20E-04	VLX-HI-SA
2	3.05E-04	26.41	FAILURE OF THE 12 VAC DISTRIBUTION PANEL	3.05E-04	EDSMOD01
3	1.68E-04	14.55	COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS ACT.)	5.06E-01	REC-MANDASC
			OPERATOR FAILS TO RECOGNIZE NEED AND FAILS TO START HYDROGEN CONTROL SYSTEM	3.32E-04	VLN-MAN01
4	7.58E-05	6.56	CCF OF HYDROGEN ANALYZER SENSORS	7.58E-05	VLX-ANLYZ
5	4.24E-05	3.67	TRANSFORMER, STATIC XFER SW FAIL TO SW, OR CKT BKR OPENS	1.57E-02	EDSMOD12
			UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS001TM
6	4.24E-05	3.67	TRANSFORMER, STATIC XFER SW FAIL TO SW, OR CKT BKR OPENS	1.57E-02	EDSMOD12
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS013TM
7	4.24E-05	3.67	TRANSFORMER, STATIC XFER SW FAIL TO SW, OR CKT BKR OPENS	1.57E-02	EDSMOD12
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS132TM
8	1.07E-05	0.93	STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO1DG001TM
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
			STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO2DG002TM
9	1.02E-05	0.88	CCF TO START OF ENGINE-DRIVEN FUEL PUMPS	2.00E-03	ZOX-PD-ES
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
10	7.29E-06	0.63	UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC1BS001TM
			UNAVAILABILITY OF BUS ECS ES 2 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC2BS002TM

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
11	7.29E-06	0.63	UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC1BS001TM
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS023TM
12	7.29E-06	0.63	UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC1BS001TM
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS232TM
13	7.29E-06	0.63	BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS013TM
			UNAVAILABILITY OF BUS ECS ES 2 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC2BS002TM
14	7.29E-06	0.63	BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS013TM
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS023TM
15	7.29E-06	0.63	BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS013TM
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS232TM
16	7.29E-06	0.63	BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS132TM
			UNAVAILABILITY OF BUS ECS ES 2 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC2BS002TM
17	7.29E-06	0.63	BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS132TM
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS023TM
18	7.29E-06	0.63	BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS132TM
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS232TM
19	4.72E-06	0.41	D/G FAILS TO START & RUN OR BKR 102 FAILS TO CLOSE	2.02E-02	ZO1MOD01
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
			STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO2DG002TM
20	4.72E-06	0.41	STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO1DG001TM
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
			D/G FAILS TO START & RUN OR BKR 202 FAILS TO CLOSE	2.02E-02	ZO2MOD01

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
21	3.71E-06	0.32	COMMON CAUSE FAILURE 4KV BREAKER TO CLOSE	7.30E-04	ECX-CB-GC
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
22	3.67E-06	0.32	TRANSFORMER, STATIC XFER SW FAIL TO SW, OR CKT BKR OPENS	1.57E-02	EDSMOD12
			STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO1DG001TM
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
23	3.32E-06	0.29	FAILURE OF MANUAL DAS REACTOR TRIP HARDWARE	1.00E-02	MDAS
			OPERATOR FAILS TO RECOGNIZE NEED AND FAILS TO START HYDROGEN CONTROL SYSTEM	3.32E-04	VLN-MAN01
24	2.24E-06	0.19	COMMON CAUSE FAILURE STANDBY DG TO RUN	4.40E-04	ZOX-DG-DR
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
25	2.13E-06	0.18	COMMON CAUSE FAILURE 4KV BREAKERS TO OPEN	4.20E-04	ECX-CB-GO
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
26	2.07E-06	0.18	D/G FAILS TO START & RUN OR BKR 102 FAILS TO CLOSE	2.02E-02	ZO1MOD01
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
			D/G FAILS TO START & RUN OR BKR 202 FAILS TO CLOSE	2.02E-02	ZO2MOD01
27	1.64E-06	0.14	FAILURE OF MANUAL DAS ACT.	1.16E-02	REC-MANDAS
			CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PL3MOD1
28	1.61E-06	0.14	TRANSFORMER, STATIC XFER SW FAIL TO SW, OR CKT BKR OPENS	1.57E-02	EDSMOD12
			D/G FAILS TO START & RUN OR BKR 102 FAILS TO CLOSE	2.02E-02	ZO1MOD01
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
29	1.42E-06	0.12	COMMON CAUSE FAILURE STANDBY DG TO START	2.80E-04	ZOX-DG-DS
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
30	1.41E-06	0.12	FAILURE OF MANUAL DAS REACTOR TRIP HARDWARE	1.00E-02	MDAS
			CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PL3MOD1
31	1.12E-06	0.1	FAILURE OF MANUAL DAS ACT.	1.16E-02	REC-MANDAS
			CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PL303
32	1.00E-06	0.09	INDICATION FAILURE	1.00E-06	ALL-IND-FAIL
33	9.81E-07	0.08	BREAKER 100 FAILS TO OPEN [#3,5]	4.20E-03	EC1CB100VO
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
			STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO2DG002TM
34	9.81E-07	0.08	STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO1DG001TM
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01
			BREAKER 200 FAILS TO OPEN [#3,5]	4.20E-03	EC2CB200VO
35	9.69E-07	0.08	FAILURE OF MANUAL DAS REACTOR TRIP HARDWARE	1.00E-02	MDAS
			CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PL303
36	7.54E-07	0.07	TRANSFORMER, STATIC XFER SW FAIL TO SW, OR CKT BKR OPENS	1.57E-02	EDSMOD12
			FIXED COMPONENT FAULTS	4.80E-05	EC1MOD13
37	6.78E-07	0.06	TRANSFORMER, STATIC XFER SW SPUR FAIL, OR CKT BKR OPENS	4.32E-05	EDSMOD11
			TRANSFORMER, STATIC XFER SW FAIL TO SW, OR CKT BKR OPENS	1.57E-02	EDSMOD12
38	6.71E-07	0.06	STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO1DG001TM
			FIXED COMPONENTS FAILURE	3.17E-04	ED4MOD112
			STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO2DG002TM

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
39	6.71E-07	0.06	STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO1DG001TM
			FIXED COMPONENTS FAILURE	3.17E-04	ED4MOD11
			STANDBY DG UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.60E-02	ZO2DG002TM
40	6.60E-07	0.06	CCF TO RUN OF ENGINE-DRIVEN FUEL PUMPS	1.30E-04	ZOX-PD-ER
			MAIN GEN. BKR ES 01 FAILS TO OPEN [# 12]	5.08E-03	EC0MOD01

Table 1a. AP1000 PRA REACTOR CAVITY FLOODING CUTSETS

NUMBER	CUTSET PROB	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1		Deleted		
2	3.40E-03	FAILURE TO RECOGNIZE THE NEED AND FAILURE TO OPEN THE RECIRCULATION VALVES TO FLOOD REACTOR CAVITY AFTER CORE DAMAGE	3.40E-03	REN-MAN03
3	5.80E-05	CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
4	1.20E-05	CCF OF STRAINERS IN IRWST TANK	1.20E-05	IWX-FL-GP
5	1.10E-05	CCF OF PMS ESF OUTPUT LOGIC SOFTWARE	1.10E-05	CCX-PMXMOD1-SW
6	8.62E-06	CCF OF EPO BOARDS IN PMS	8.62E-06	CCX-EP-SAM
7		Deleted		
8	2.13E-06	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
		HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
9	1.28E-06	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
		RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
10	1.28E-06	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
		HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
11	1.20E-06	SOFTWARE CCF OF ALL CARDS	1.20E-06	CCX-SFTW
12	7.67E-07	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
		RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA

NUMBER	CUTSET PROB	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
13	4.38E-07	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDS1TM
14	4.38E-07	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDD1TM
15	4.38E-07	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
16	4.38E-07	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
17	3.50E-07	SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
		HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
18	3.50E-07	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
		SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
19	2.63E-07	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
20	2.63E-07	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
21	2.63E-07	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDS1TM
22	2.63E-07	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDD1TM
23	2.50E-07	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11

NUMBER	CUTSET PROB	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
		FAILURE OF OUTPUT DRIVER	1.71E-04	IRCEP118ASA
24	2.50E-07	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
		FAILURE OF THE POWER INTERFACE BOARD (###EP####SA)	1.71E-04	IRDEP118BSA
25	2.10E-07	SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
		RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
26	2.10E-07	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
		SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
27	2.06E-07	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
		CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PMBMOD1
28	2.06E-07	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
		CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PMAMOD1
29	1.50E-07	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
		FAILURE OF OUTPUT DRIVER	1.71E-04	IRCEP118ASA
30	1.50E-07	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
		FAILURE OF THE POWER INTERFACE BOARD (###EP####SA)	1.71E-04	IRDEP118BSA
31	1.41E-07	HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
		CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PMB030
32	1.41E-07	HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
		CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PMA030
33	1.24E-07	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
		CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PMBMOD1

NUMBER	CUTSET PROB	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
34	1.24E-07	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
		CCF OF OUTPUT LOGIC I/Os (CCX- P##MOD1)	1.41E-04	CCX-PMAMOD1
35	9.00E-08	BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDS1TM
36	9.00E-08	BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDD1TM
37	9.00E-08	BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDS1TM
38	9.00E-08	BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
		BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDD1TM
39	8.49E-08	RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
		CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PMB030
40	8.49E-08	RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
		CCF OF THE LOGIC GROUP PROCESSING (CCX-###03)	9.69E-05	CCX-PMA030

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.2.3.3-1 Revision 2

Original RAI Number(s): None

Summary of Issue:

The AP1000 insulation design was refined based on insights from the Configuration IV tests, and a prototypical insulation design for AP1000 was evaluated as part of the ULPU Configuration V test program. The applicant has indicated that the Configuration V test results show a further improvement in coolability performance relative to Configuration IV, and also include information on transient pressure loads needed by the COL-applicant to establish the pressure loads for the structural analysis of the final insulation design. The applicant has not provided documentation of: the RPV insulation design evaluated in Configuration V, the results of the Configuration V testing, or the functional requirements for the AP1000 RPV insulation system. Such information is needed in order for the staff to conclude on the margins to lower head failure for AP1000, and the viability of Westinghouse's proposal that the COL applicant complete the RPV insulation design. This is Open Item 19.2.3.3-1.

Westinghouse Response:

Attachment 3 to Westinghouse letter DCP/NRC1603 dated July 8, 2003 provides the ULPU V test report that can be used by the COL applicant to complete the RPV insulation design. The in-vessel retention functional requirements for the RPV insulation design are given in the AP1000 PRA Section 39.10.2. The pressure data from the ULPU V testing will be used by the COL applicant to determine loads on the insulation and its supporting structure. The ULPU V test results indicate that the pressure variations in the flow channel between the vessel and the insulation are on the order of plus/minus 0.5 meters of water. Fast Fourier Transform analysis of the ULPU V pressure data is also included in the ULPU V test report. This analysis shows that the dominant frequency of the pressure variations is less than about 2 Hz. The natural frequency of the insulation structure is expected to be well above 2 Hz, so the observed pressure variations will most likely be treated as static pressure loads in the design of the insulation structure.

DCD Subsection 5.3.5.4 will be revised as shown below.

Design Control Document (DCD) Revision:

5.3.5.4 Determination of Forces on Insulation and Support System

The forces that may be expected in the reactor cavity region of the AP1000 plant during a core damage accident in which the core has relocated to the lower head and the reactor cavity is reflooded can be based on test results from the ULPU test program (Reference 5). The particular configuration (Configuration V) reviewed closely models the full-scale AP1000 geometry of water in the region near the reactor vessel, between the reactor vessel and the reactor vessel insulation. The ULPU tests provide data on the pressure generated in the region between the reactor vessel and reactor vessel insulation. These data, along with observations

and conclusions from heat transfer studies, are used to develop the functional requirements with respect to in-vessel retention for the reactor vessel insulation and support system. Interpretation of data collected from ULPU Configuration V experiments in conjunction with the static head of water that would be present in the AP1000 is used to estimate forces acting on the rigid sections of insulation. The ULPU V test results indicate that the pressure variations in the flow channel between the vessel and the insulation are on the order of plus/minus 0.5 meters of water. Fast Fourier Transform analysis of the ULPU V pressure data is also included in the ULPU V test report. This analysis shows that the dominant frequency of the pressure variations is less than about 2 Hz. The natural frequency of the insulation structure is expected to be well above 2 Hz.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.4-1 Response Revision 2

Original RAI Number(s): 720.060

Summary of Issue:

In a revised RAI response dated March 31, 2003, the applicant provided an updated evaluation addressing these concerns. The staff has not completed of its evaluation of SAMDAs for AP1000. Therefore, this is Open Item 19.4-I.

Westinghouse Response:

Westinghouse believes that the response to RAI 720.060 revision 1 dated March 31, 2003 provides a revised SAMDA evaluation that complies with NRC concerns.

NRC Follow-on Comment:

In a teleconference held with Westinghouse, the NRC staff asked that Westinghouse provide an explanation of why a redesign of the accumulators or 4th stage ADS valves was adopted as part of the SAMDA evaluation.

Westinghouse Response:

As acknowledged by the NRC in the teleconference, the very low AP1000 risk profile is such that the perfect SAMDA (i.e. one that totally eliminates offsite consequences) would have to cost less than $33,000 to meet the risk worth necessary to be considered. The following addresses the two items that were raised in the teleconference by the NRC.

Larger accumulators

Increasing the size of the accumulators would result in a significant increase in cost that would be greater than the cost threshold established by the perfect SAMDA evaluation in our earlier response. In order to have any benefit in the PRA, the accumulators would have to be increased in size sufficiently to change the Large LOCA success criteria from 2 of 2 accumulators to 1 of 2 accumulators. Westinghouse estimates that the accumulator tanks would have to be increased in size from 2000ft^3 to 4000 ft^3, and the hardware costs associated with this change would be significant. Such a size increase would also likely result in a change to the design of the DVI piping subsystem. The design of this piping system was established in the AP600 design certification, and the design does not change significantly for AP1000. Recently Westinghouse completed the leak-before break analysis of the DVI piping, and any change in the DVI piping would result in significant piping reanalysis of the DVI piping. Westinghouse estimates the redesign costs associated with the changes in hardware and piping re-design to be significantly greater than the cost threshold established for the perfect SAMDA discussed in our earlier SAMDA evaluation. Therefore this design change was not incorporated.

Larger 4th stage ADS valves

Increasing the 4th stage ADS valves in size would result in a significant increase in cost associated with redesigning the AP1000 loop piping and 4th stage piping configuration. The AP1000 ADS valves were already increased in size compared to the AP600 valves more than the ratio of the power uprate of the AP1000. In order to have any benefit in the PRA, the 4th stage ADS valves would have to be increased in size sufficiently to change the LOCA success criteria from 3 of 4 valves to 2 of 4 valves. To accommodate such a change, Westinghouse estimates that the 4th stage ADS valves would have to increase in size from 14-inch to 18-inch valves and associated piping. In addition, the common 4th stage inlet piping that connects to the hot leg would have to increase in size from 18-inch to at least 20-inch. This would require a significant redesign of the squib valve, and would also result in re-design of theADS-4 piping which in-turn would impact the design of the reactor coolant loop piping. Finally, such a redesign would require Westinghouse to perform additional confirmatory testing of the passive core cooling system to verify that the behavior of the passive safety systems was not adversely impacted. Westinghouse estimates the cost of this change to be significantly larger than the cost threshold of the perfect SAMDA established in our earlier response. Therefore, this design change was not incorporated.

NRC Follow-on Comment:

The information provided in response to RAI 720.060 and in the 10/6/03 response to this open item should be added to the DCD.

Westinghouse Response

AP1000 DCD Appendix 1B is revised as shown below to incorporate the previous response information.

Design Control Document (DCD) Revision:

Appendix 1B of the DCD is completely revised as shown on the following pages.

PRA Revision:

None

APPENDIX 1B

SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

1B.1 AP1000 SAMDA Evaluation

1B.1.1 Introduction

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

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

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

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

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

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

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

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

1B.1.2 Summary

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

radiological risk profile of the AP600 design. Since the AP1000 is so similar to the AP600, the list of candidate modifications is the same.

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

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

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

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

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

1B.1.3 Selection and Description of SAMDAs

Candidate design alternatives were selected based upon design alternatives evaluated for other plant designs (References 4, 5, and 6) as well as suggestions from AP600 and AP1000 design personnel. Additional candidate design alternatives were selected based upon an assessment of the AP600 and AP1000 probabilistic risk assessment results. Fifteen design alternatives were finally selected for further evaluation. These 15 SAMDAs are as follows:

- Chemical, volume, and control system (CVS) upgraded to mitigate small loss-of-coolant accidents (LOCAs)
- Filtered containment vent
- Normal residual heat removal system (RNS) located inside containment

- Self-actuating containment isolation valves
- Passive containment spray
- Active high-pressure safety injection system
- Steam generator shell-side passive heat removal system
- Steam generator safety valve flow directed to in-containment refueling water storage tank (IRWST)
- Increase of steam generator secondary side pressure capacity
- Secondary containment filtered ventilation
- Diverse IRWST injection valves
- Diverse containment recirculation valves
- Ex-vessel core catcher
- High-pressure containment design
- Diverse actuation system improved reliability.

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

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

Upgrade Chemical, Volume, and Control System for Small LOCAs

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

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

Filtered Vent

This SAMDA consists of placing a filtered containment vent and all associated piping and penetrations into the AP1000 containment design. The filtered vent could be used to vent the containment to prevent catastrophic overpressure failure, and it also provides filtering capability for source term release. With respect to the AP1000 Probabilistic Risk Assessment, the possible scenario in which the filtered vent could result in risk reduction would be late containment overpressure failures (release category CFL). Other containment overpressure failures occur due to dynamic severe accident phenomena, such as hydrogen burn and steam explosion. The late containment failures for AP1000 are failures of the passive containment cooling system. Analyses have indicated that for scenarios with passive containment cooling system failure, air cooling may limit the containment pressure to less than the ultimate pressure. However, for the Level 2 probabilistic risk assessment, failure of the passive containment cooling system is assumed to result in containment failure based on an adiabatic heatup. To conservatively consider the risk reduction of a filtered vent, the use of a filtered vent to preclude a late containment failure will be evaluated. A decontamination factor (DF) of 1000 will conservatively be assumed for each probabilistic risk assessment Level 1 accident classification, even though it is realized that the dose due to noble gases will not be impacted by the filtered vent since 100 percent of the noble gas fission products will still be released. Therefore, the risk reduction is equal to the decontamination factor assumed since the probabilistic risk assessment Level 1 accident classification frequencies do not change.

Self-Actuating Containment Isolation Valves

This SAMDA consists of improved containment isolation provisions on all normally open containment penetrations. The category of "normally open" is limited to normally open pathways to the environment during power and shutdown conditions, excluding closed systems inside and outside the containment such as normal residual heat removal system and component cooling. The design alternative would be to add a self-actuating valve or enhance the existing inside containment isolation valve to provide for self-actuation in the event that containment conditions are indicative of a severe accident. Conceptually, the design would be either an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment

conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve. This provides the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures that occur at the time of the accident, such as in cases of vessel rupture or anticipated transients without scram.

Passive Containment Sprays

This SAMDA involves adding a passive safety-related spray system and all associated piping and support systems to the AP1000 containment. A passive containment spray system could result in risk benefits in the following ways:

- Scrubbing of fission products could be done primarily for CI failures.
- Assuming appropriate timing, containment spray could be used as an alternate means for flooding the reactor vessel (in-vessel retention) and for debris quenching should vessel failure occur.
- Containment spray could also be used to control containment pressure for cases in which passive containment cooling system has failed.

In order to envelop these potential risk benefits, the risk reduction evaluation will assume that containment sprays are perfectly effective for each of these benefits, with the exception of fission product scrubbing for containment bypass. Thus, the risk reduction can be conservatively estimated by assuming all release categories except BP are eliminated.

Active High-Pressure Safety Injection System

This SAMDA consists of adding a safety-related active high-pressure safety injection pump and all associated piping and support systems to the AP1000 design. A perfect high-pressure safety injection system is assumed to prevent core melt for all events but excessive LOCA and anticipated transients without scram. Therefore, to estimate the risk reduction, only the contributions to each release category of Level 1 accident classes 3C (vessel rupture) and 3A (anticipated transients without scram) need to be considered. This SAMDA would completely change the design approach from a plant with passive safety systems to a plant with passive plus active safety-related systems, and it is not consistent with design objectives.

Steam Generator Shell-Side Heat Removal System

This SAMDA consists of providing a passive safety-related heat removal system to the secondary side of the steam generators. The system would provide closed loop cooling of the secondary using natural circulation and stored water cooling. This prevents a loss of primary heat sink in the event of a loss of startup feedwater and passive residual heat removal heat exchanger. A perfect secondary heat removal system would eliminate transients from each of the release categories. In order to evaluate the benefit of this SAMDA, the frequencies of all the transient sequences are subtracted from the overall frequency of each of the release categories and the risk is recalculated.

Direct Steam Generator Relief Flow to the In-containment Refueling Water Storage Tank

This SAMDA consists of providing all the piping and valves required for redirecting the flow from the steam generator safety and relief valves to the IRWST. An alternate, lower cost option of this SAMDA consists of redirecting only the first-stage safety valve to the IRWST. This system would prevent or reduce fission product release from bypassing the containment in the event of a steam generator tube rupture event. In order to evaluate the benefit from this SAMDA (both options), this design change is assumed to eliminate the BP release category.

Increased Steam Generator Pressure Capability

This SAMDA consists of increasing the design pressure of the steam generator secondary side and safety valve set point to the degree that a steam generator tube rupture will not cause the secondary system safety valve to open. The design pressure would have to be increased sufficiently such that the combined heat capacity of the secondary system inventory and the passive residual heat removal system could reduce the reactor coolant system temperature below T_{sat} for the secondary design pressure. Although specific analysis would have to be performed, it is estimated that the design pressure would have to be increased several hundred psi. This design would also prevent the release of fission products that bypass the containment via the steam generator tube rupture.

Secondary Containment Filtered Ventilation

This SAMDA consists of providing the middle and lower annulus (below the 135'-3' elevation) of the secondary concrete containment with a passive annulus filter system to for filtration of elevated releases. The passive filter system is operated by drawing a partial vacuum on the middle annulus through charcoal and HEPA filters. The partial vacuum is drawn by an eductor with motive flow from compressed gas tanks. The secondary containment would then reduce particulate fission product release from any failed containment penetrations (containment isolation failure). In order to evaluate the benefit from such a system, this design change is assumed to eliminate the CI release category.

Diverse In-containment Refueling Water Storage Tank Injection Valves

This SAMDA consists of changing the IRWST injection valve designs so that two of the four lines use diverse valves. Each of the four lines is currently isolated by a squib valve in series with a check valve. In order to provide diversity, the valves in two of the lines will be provided by a different vendor. For the check valves, alternate vendors are available. However, it is questionable if check valves of different vendors would be sufficiently different to be considered diverse unless the type of check valve was changed from the current swing disk check to another type. The swing disk type is the preferred type for this application and other types are considered to be less reliable. Squib valves are specialized valve designs for which there are few vendors. A vendor may not be willing to design, qualify, and build a reasonable squib valve design for this AP1000 application considering that they would only supply two valves per plant. As a result, this SAMDA is not really practicable because of the uncertainty in availability of a second squib valve design/vendor and because of the uncertainty in the reliability of another check valve type. However, the cost estimate for this SAMDA assumes that a second squib valve vendor exists and

that the vendor provides only the two diverse IRWST squib valves. The cost impact does not include the additional first time engineering and qualification testing that will be incurred by the second vendor. Those costs are expected to be more than a million dollars.

This change will reduce the frequency of core melt by eliminating the common cause failure of the IRWST injection. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of IRWST injection are assumed to be averted. Core damage sequences resulting from a failure of IRWST injection correspond to probabilistic risk assessment Level 1 accident classification 3BE; thus, release category 3BE is eliminated.

Diverse Containment Recirculation Valves

This SAMDA consists of changing the containment recirculation valve designs so that two out of the four lines use diverse valves. Each of the four lines currently contains a squib valve; two of the lines contain check valves, and the other two contain motor-operated valves. In order to provide diversity, the squib valves in two lines will be made diverse by supplying them from a different vendor. This change will reduce the frequency of core melt by eliminating the common cause failure of the containment recirculation. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of containment recirculation are assumed to be averted. Core damage sequences resulting from failure of containment recirculation correspond to probabilistic risk assessment Level 1 accident classification 3BL; thus, release category 3BL is eliminated.

Ex-Vessel Core Catcher

This SAMDA consists of designing a structure in the containment cavity or using a special concrete or coating that will inhibit core-concrete interaction (CCI), even if the debris bed dries out. A perfect core catcher would prevent CCI for all cases. However, the AP1000 incorporates a wet cavity design in which ex-vessel cooling is used to maintain the core debris in the vessel to prevent ex-vessel phenomena, such as CCI. Consequently, containment failure due to CCI is not considered in detail for the AP1000 Level 2 probabilistic risk assessment. For cases in which reactor vessel flooding is failed, it is assumed that containment failure occurs due to ex-vessel steam explosion or CCI. This containment failure is assumed to be an early containment failure, CFE (due to ex-vessel steam explosion) even though CCI and basemat melt-through would be a late containment failure. To conservatively estimate the risk reduction of an ex-vessel core catcher, this design change is assumed to eliminate the CFE release category.

High-Pressure Containment Design

This SAMDA design consists of using the massive high-pressure containment design in which the design pressure of the containment is approximately 300 psi (20 bar) for the AP1000 containment. The massive containment design has a passive containment cooling feature much like the AP1000 containment. The high design pressure is considered only for prevention of containment failures due to severe accident phenomena, such as steam explosions and hydrogen detonation. A perfect high-pressure containment design would reduce the probability of containment failures, but would have no reduction of the frequency or magnitude of the release from an unisolated containment (containment isolation failure or containment bypass). To estimate

the risk reduction of a high-pressure containment design, this design is assumed to eliminate the CFE, CFI, and CFL release categories.

Increase Reliability of Diverse Actuation System

This SAMDA design consists of improving the reliability of the diverse actuation system, which actuates engineered safety features and allows the operator to monitor the plant status. The design change would add a third instrumentation and control cabinet and a third set of diverse actuation system instruments to allow the use of two-out-of-three logic instead of two-out-of-two logic. Other changes, such as adding another set of batteries, have not been included in the cost estimates. A perfectly reliable diverse actuation system would reduce the frequency of the release categories by the cumulative frequencies of all sequences in which diverse actuation system failure leads to core damage. In order to evaluate the benefit from the diverse actuation system upgrade, a Level 1 sensitivity analysis assuming perfect reliability of diverse actuation system was completed.

Locate Normal Residual Heat Removal Inside Containment

This SAMDA consists of placing the entire normal residual heat removal system and piping inside the containment pressure boundary. Locating the normal residual heat removal system inside the containment would prevent containment bypass due to interfacing system LOCAs (ISLOCA) of the residual heat removal system. In past probabilistic risk assessments of current generation nuclear power plants, the ISLOCA is the leading contributor of plant risk because of large offsite consequences. A failure of the valves which isolate the low-pressure residual heat removal system from the high pressure reactor coolant system causes the residual heat removal system to overpressurize and fail, releasing reactor coolant system coolant outside the containment where it cannot be recovered for recirculation cooling of the core. The result is core damage and the direct release of fission products outside the containment.

In the AP1000, the normal residual heat removal system is designed with a higher design pressure than the systems in current pressurized water reactors, and an additional isolation valve is provided in the design. In the probabilistic risk assessment, no ISLOCAs contribute significantly to the core damage frequency (CDF) of the AP1000 (Reference 2, Chapter 33). Therefore, relocating the normal residual heat removal system of the AP1000 inside containment will provide virtually no risk reduction benefit and will not be investigated further in terms of cost.

1B.1.4 Methodology

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

1B.1.4.1 Total Population Dose

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

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

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

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

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

The release categories for the AP1000 are defined as follows:

- IC – intact containment. Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.
- CFE – containment failure early. Fission-product release through a containment failure caused by severe accident phenomenon occurring after the onset of core damage but prior to core relocation.
- CFI – containment failure intermediate. Fission-product release through a containment failure caused by severe accident phenomenon occurring after core relocation but before 24 hours.
- CFL – containment failure late. Fission-product release through a containment failure caused by severe accident phenomenon occurring after 24 hours.
- CI – containment isolation failure. Fission-product release through a failure of the system or valves that close the penetrations between the containment and the environment. Containment failure occurs prior to onset of core damage.
- BP – containment bypass. Fission products are released directly from the Reactor Coolant System to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.

The following subsections present a brief description of the AP1000 release categories.

Release Category IC – Intact Containment

If the containment integrity is maintained throughout the accident, then the release of radiation from the containment is due to nominal leakage and is expected to be within the design basis of the containment. This is the "no failure" containment failure mode and is termed intact containment. The main location for fission-product leakage from the containment is penetration leakage into the auxiliary building where significant deposition of aerosol fission products may occur.

Release Category CFE – Early Containment Failure

Early containment failure is defined as failure that occurs in the time frame between the onset of core damage and the end of core relocation. During the core melt and relocation process, several dynamic phenomena can be postulated to result in rapid pressurization of the containment to the point of failure. The combustion of hydrogen generated in-vessel, steam explosions, and reactor vessel failure from high pressure are major phenomena postulated to have the potential to fail the containment. If the containment fails during or soon after the time when the fuel is overheating and starting to melt, the potential for attenuation of the fission-product release diminishes because of short fission-product residence time in the containment. The fission products released to the containment prior to the containment failure are discharged at high pressure to the environment as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Containment failures postulated within the time of core relocation are binned into release category CFE.

Release Category CFI – Intermediate Containment Failure

Intermediate containment failure is defined as failure that occurs in the time frame between the end of core relocation and 24 hours after core damage. After the end of the in-vessel fission-product release, the airborne aerosol fission products in the containment have several hours for deposition to attenuate the source term. The global combustion of hydrogen generated in-vessel from a random ignition prior to 24 hours can be postulated to fail the containment. The fission products in the containment atmosphere are discharged at high pressure to the environment as the containment blows down. Containment failures postulated within 24 hours of the onset of core damage are binned into release category CFI.

Release Category CFL – Late Containment Failure

Late containment failure is defined as containment failure postulated to occur later than 24 hours after the onset of core damage. Since the probabilistic risk assessment assumes the dynamic phenomena, such as hydrogen combustion, to occur before 24 hours, this failure mode occurs only from the loss of containment heat removal via failure of the passive containment cooling system. The fission products that are airborne at the time of containment failure will be discharged at high pressure to the environment, as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Accident sequences with failure of containment heat removal are binned in release category CFL.

Release Category CI – Containment Isolation Failure

A containment isolation failure occurs because of the postulated failure of the system or valves that close the penetrations between the containment and the environment. Containment isolation failure occurs before the onset of core damage. For such a failure, fission-product releases from the reactor coolant system can leak directly from the containment to the environment with diminished potential for attenuation. Most isolation failures occur at a penetration that connects the containment with the auxiliary building. The auxiliary building may provide additional attenuation of aerosol fission-product releases. However, this decontamination is not credited in the containment isolation failure cases. Accident sequences in which the containment does not isolate prior to core damage are binned into release category CI.

Release Category BP – Containment Bypass

Accident sequences in which fission products are released directly from the reactor coolant system to the environment via the secondary system or other interfacing system bypass the containment. The containment failure occurs before the onset of core damage and is a result of the initiating event or adverse conditions occurring at core uncovery. The fission-product release to the environment begins approximately at the onset of fuel damage, and there is no attenuation of the magnitude of the source term from natural deposition processes beyond that which occurs in the reactor coolant system, in the secondary system, or in the interfacing system. Accident sequences that bypass the containment are binned into release category BP.

1B.1.4.2 AP1000 Risk (CDF, LRF, and POPULATION Dose)

Table 1B-2 presents a summary of the CDF and large release frequency (LRF) risks for the AP1000.

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

1B.1.5 Summary of Risk Significant Enhancements

This section summarizes the design enhancements already incorporated into the AP1000 plant due to probabilistic risk assessment insights and results.

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

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

- Changed IRWST drain procedure so it occurs earlier for IVR support

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

- Improved IVR heat transfer

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

- Improved IRWST vents

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

- Incorporated low boron core (anticipated transients without scram)

  In the AP600, anticipated transients without scram (ATWS) contribution to LRF was noticed to be high relative to other initiating events. A low boron core was incorporated into the design to reduce the potential contribution of ATWS to plant risk.

- Added 3rd passive containment cooling drain valve (motor-operator valve diverse to air-operated valve)

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

- Reduced potential recirculation-line squib valve failures

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

1B.1.6 Specific Site Characteristics

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

1B.1.7 Value of Eliminating Risk

The dollar value of completely eliminating all severe accident risk for an AP1000 plant at the reference site is calculated below for a base case, and various sensitivity analyses.

The following cost categories are considered:

• Public exposure	$2000 per man-rem
• Loss of plant	$2.0E+09
• Offsite property damage/cleanup	$2.0E+09
• Onsite cleanup and decontamination	$1.0E+09
• Replacement power	not considered since the plant is written off

NUREG/CR-3568 ("A Handbook for Value-Impact Assessment," 1983) is consulted for setting up the base case.

The following additional input are used for the estimate:

• Delta CDF	2.41E-07/yr
• Delta LRF	1.95E+08/yr
• Average population whole body dose	6.4E+05 man-rem
• Plant life	40 years

The inflation rate is taken as equal to opportunity cost of money. This is conservative in most cases since generally, the opportunity cost of money is larger than inflation, which makes the current value of a series of future expenditures less than a simple sum of all these expenditures.

The calculations for the base case and three more conservative cases are shown in Table 1B-4.

The following equations are used for calculating dollar value of eliminating risk (with the assumption that inflation rate is equal to opportunity cost of money):

$Q = Q1 + Q2 + Q3 + Q4$
$Q1 = q1 * f2 * r * t$
$Q2 = q2 * f1 * t$
$Q3 = q3 * f2 * t$
$Q4 = q4 * f1 * t$

The symbols are defined in Table 1B-4.

From Table 1B-4, it is seen that, even with generously conservative assumptions, the value of eliminating AP1000 risk totally is small. The value of the ideal SAMDA is approximately $30,000. Even if the AP1000 CDF and LRF were a factor of 10 higher, this value is only $405,000.

1B.1.8 Evaluation of Potential Improvements

The value of eliminating AP1000 total risk is $30,000, as discussed in Section 1B.1.7. This value is an upper bound for any single engineered design alternative, which would actually reduce CDF and/or LRF a fraction of the values assumed in the base case for calculating the $30,000 value. Moreover, only 2 percent of the $30,000 comes from reduction of man-rem exposure. Thus, any design alternative that does not reduce CDF considerably, even if it does reduce the man-rem exposure, would not be cost beneficial.

For the AP600, 14 design alternatives discussed in this section were found to be not cost effective. One of these alternatives is actually implemented in the AP1000 design (diverse containment recirculation squib valves). The costs associated with the remaining 13 design alternatives are provided in Table 1B-5. Only one design alternative, 3 – namely, self-actuating containment isolation valves – has a cost near $30,000; the remaining alternatives are at least an order of magnitude more costly than $30,000. Thus, only design alternative 3 needs to be further discussed.

1B.1.8.1 Self-Actuating Containment Isolation Valves

This SAMDA consists of improved containment isolation provisions on all normally open containment penetrations. The category of "normally open" is limited to normally open pathways to the environment during power and shutdown conditions, excluding closed systems inside and outside the containment such as normal residual heat removal system and component cooling. The design alternative would be to add a self-actuating valve or enhance the existing inside containment isolation valve to provide for self-actuation in the event that containment conditions are indicative of a severe accident. Conceptually, the design would either be an independent valve

or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve. This provides the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures, which occur at the time of the accident such as in cases of vessel rupture or ATWS. This design alternative provides almost no benefit in reducing plant CDF.

Generously assuming that this design alternative will eliminate CI release totally, the delta LRF is 1.33E-09/yr (see Table 1B-6). Delta CDF is zero. The benefit of this design alternative is calculated as $320 (see Table 1B-7). Even with increased CDF and LRF, this value is only $22,500. Based on these calculations, even the cheapest design alternative does not meet the benefit/cost ratio of 1.

1B.1.8.2 Other New Design Changes

Other design changes, as discussed in Section 1B.1.5, are already incorporated into the AP1000. There is no cost/benefit analysis available for those changes already incorporated.

Two additional design changes not incorporated in the AP1000 were assessed as follows:

Larger Accumulators

Increasing the size of the accumulators would result in a significant increase in cost that would be greater than the cost threshold established by the perfect SAMDA evaluation. In order to have any benefit in the probabilistic risk assessment, the accumulators would have to be increased in size sufficiently to change the large LOCA success criteria from two of two accumulators to one of two accumulators. Westinghouse estimates that the accumulator tanks would have to be increased in size from 2000 ft^3 to 4000 ft^3, and the hardware costs associated with this change would be significant. Such a size increase would also likely result in a change to the design of the DVI piping subsystem. The design of this piping system was established in the AP600 design certification, and the design does not change significantly for AP1000. Recently, Westinghouse completed the leak-before break analysis of the DVI piping, and any change in the DVI piping would result in significant piping reanalysis of the DVI piping. Westinghouse estimates the redesign costs associated with the changes in hardware and piping re-design to be significantly greater than the cost threshold established for the perfect SAMDA discussed above. Therefore this design change was not incorporated.

Larger Fourth-Stage ADS Valves

Increasing the fourth-stage ADS valves in size would result in a significant increase in cost associated with redesigning the AP1000 loop piping and fourth-stage piping configuration. The AP1000 ADS valves were already increased in size compared to the AP600 valves more than the ratio of the power uprate of the AP1000. In order to have any benefit in the probabilistic risk assessment, the 4th stage ADS valves would have to be increased in size sufficiently to change the LOCA success criteria from three of four valves to two of four valves. To accommodate such a change, Westinghouse estimates that the fourth-stage ADS valves would have to increase in size from 14-inch to 18-inch valves and associated piping. In addition, the common fourth-stage inlet

piping that connects to the hot leg would have to increase in size from 18-inch to at least 20-inch. This would require a significant redesign of the squib valve and would also result in redesign of the ADS-4 piping which in turn would impact the design of the reactor coolant loop piping. Finally, such a redesign would require Westinghouse to perform additional confirmatory testing of the passive core cooling system to verify that the behavior of the passive safety systems was not adversely impacted. Westinghouse estimates the cost of this change to be significantly larger than the cost threshold of the perfect SAMDA discussed above. Therefore, this design change was not incorporated.

1B.1.9 Results

Due to the existing low risk of the AP1000 plant, none of the design alternatives described in Section 1B.1.3 meet an acceptable benefit to cost ratio of 1 or greater.

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

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

As the AP1000 plant CDF is lower than for existing plants, the benefits of additional design alternatives are small. The 15 SAMDAs analyzed provided little or no benefit to the AP1000 design.

Assuming a hypothetical design alternative was developed which provides a 100-percent reduction in overall plant risk, representing an averted risk of 1.24×10^{-2} man-rem per year, the capital benefit amounts to only $31,500.

1B.2 References

1. "SECY-91-229 - Severe Accident Mitigation Design Alternatives for Certified Standard Designs," USNRC Memorandum from Samuel J. Chilk to James M. Taylor, dated October 25, 1991.

2. "AP1000 Probabilistic Risk Assessment", APP-GW-GL-022, Revision 5, Westinghouse Electric Company, December 2003.

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

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

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

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

7. "Technical Assessment Guide," EPRI P-6587-L, Volume 1, Revision 6, September 1989.

8. Nuclear Energy Cost Data Base, DOE/NE-0095, U.S. Department of Energy, September 1988.

9. Chanin, D., Young, M. L., "Code Manual for MACCS2, User's Guide," NUREG/CR-6613, SAND97-0594, Vol. 1, Sandia National Laboratories, U.S. Nuclear Regulatory Commission.

10. "PRA Procedures Guide," NUREG/CR-2300, U.S. Nuclear Regulatory Commission, Vol. 2, Chapter 9, Washington, D.C.

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

12. NRC NUREG/CR-2239 "Technical Guidance for Siting Criteria Development," prepared by Sandia National Laboratories, D.C. Aldrich, et al, December 1982.

Table 1B-1

POPULATION WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (per reactor year)	Mean Dose (person-sieverts)	Dose (person-REM)	Risk (person-REM per reactor year)	Percentage Contribution to Total Risk
CFI	1.89E-10	7.88E+03	7.88E+05	1.49E-04	1.2
CFE	7.47E-09	8.51E+03	8.51E+05	6.36E-03	51.3
IC	2.21E-07	7.19E+00	7.19E+02	1.59E-04	1.3
BP	1.05E-08	2.91E+03	2.91E+05	3.06E-03	24.7
CI	1.33E-09	2.01E+04	2.01E+06	2.67E-03	21.6
CFL	3.45E-13	5.32E+03	5.32E+05	1.84E-07	0.0
			Total Risk =	1.24E-02	100.0

Table 1B-2

SUMMARY OF AP1000 RISK (CDF AND LRF)

	CDF	LRF
Internal events at power	2.41E-07/yr	1.95E-08/yr
Events at shutdown	1.23E-07/yr	2.05E-08/yr (2)
Internal fire	5.61E-08/yr	4.54E-09/yr (2)
Internal flooding	8.82E-10/yr	negligible
Seismic events	not quantified (1)	not quantified (1)

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

Table 1B-3

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

24-Hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CTI	7.85E+03	6.11E+03	1.47E+04	2.01E+04	3.21E+04	3.51E+04	5.34E+04
CTE	8.51E+03	6.25E+03	1.62E+04	2.31E+04	4.13E+04	5.06E+04	6.40E+04
DIRECT	2.16E+01	1.20E+01	4.78E+01	8.13E+01	1.14E+02	1.23E+02	1.68E+02
IC	7.19E+00	4.21E+00	1.71E+01	2.95E+01	3.56E+01	3.84E+01	5.60E+01
BP	2.91E+03	1.74E+03	5.90E+03	1.00E+04	1.52E+04	1.81E+04	2.58E+04
CI	2.01E+04	1.13E+04	4.71E+04	6.60E+04	1.23E+05	1.48E+05	1.61E+05
CTL	5.32E+03	3.87E+03	1.04E+04	1.35E+04	2.32E+04	2.77E+04	4.35E+04
72-Hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CTI	8.89E+03	6.89E+03	1.63E+04	2.21E+04	3.42E+04	3.84E+04	5.73E+04
CTE	9.36E+03	6.89E+03	1.88E+04	2.54E+04	4.25E+04	5.12E+04	6.77E+04
DIRECT	2.45E+01	1.43E+01	5.50E+01	8.33E+01	1.16E+02	1.26E+02	1.78E+02
IC	8.80E+00	5.57E+00	1.98E+01	3.14E+01	4.41E+01	5.03E+01	6.33E+01
BP	3.11E+03	1.85E+03	6.31E+03	1.03E+04	1.54E+04	1.82E+04	2.69E+04
CI	2.14E+04	1.25E+04	4.90E+04	7.40E+04	1.27E+05	1.53E+05	1.67E+05
CTL	5.84E+03	4.32E+03	1.12E+04	1.48E+04	2.53E+04	3.04E+04	4.62E+04

Table 1B-4

VALUE OF ELIMINATING RISK

		Base Case	Case 2
f1	Delta CDF	2.41E-07	2.41E-06
f2	Delta LRF	1.95E-08	1.95E-07
r	Man-REM exposure	6.40E+05	6.40E+06
t	Plant life	4.00E+01	4.00E+01
q1	Cost of exposure	$2,000	$2,000
q2	Cost of plant	$2,000,000,000	$2,000,000,000
q3	Offsite damage	$2,000,000,000	$2,000,000,000
q4	Onsite cleanup	$1,000,000,000	$1,000,000,000
Q1	Value of exposure =	$1,154	$115,440
Q2	Value of plant =	$19,280	$192,800
Q3	Value of offsite damage =	$1,560	$15,600
Q4	Value of onsite cleanup =	$9,640	$96,400
Q	Total value of eliminating risk =	$31,478	$404,640

Table 1B-5

DESIGN ALTERNATIVES FOR SAMDA

No.	Design Alternative	Cost
1	Upgrade chemical, volume, and control system for small LOCA	1,500,000
2	Containment filtered vent	5,000,000
3	Self-actuating containment isolation valves	33,000
4	Safety grade passive containment spray	3,900,000
6	Steam generator shell-side heat removal	1,300,000
7	Steam generator relief flow to IRWST	620,000
8	Increased steam generator pressure capability	8,200,000
9	Secondary containment ventilation with filtration	2,200,000
10	Diverse IRWST injection valves	570,000
11	Diverse containment recirculation valves	Already Implemented
12	Ex-vessel core catcher	1,660,000
13	High-pressure containment design	50,000,000
14	More reliable diverse actuation system	470,000

Table 1B-6

POPULATION WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (per reactor year)	Mean Dose (person-sieverts)	Dose (person-REM)	Risk (person-REM per reactor year)	Percentage Contribution to Total Risk
CFI	1.89E-10	7.88E+03	7.88E+05	1.49E-04	1.2
CFE	7.47E-09	8.51E+03	8.51E+05	6.36E-03	51.3
IC	2.21E-07	7.19E+00	7.19E+02	1.59E-04	1.3
BP	1.05E-08	2.91E+03	2.91E+05	3.06E-03	24.7
CI	1.33E-09	2.01E+04	2.01E+06	2.67E-03	21.6
CFL	3.45E-13	5.32E+03	5.32E+05	1.84E-07	0.0
			Total Risk =	1.24E-02	100.0

Table 1B-7

VALUE OF ELIMINATING RISK FOR ALTERNATIVE 3

	Base Case	Case 2
Delta CDF	0.00E+00	0.00E+00
Delta LRF	1.33E-09	1.33E-08
Man-REM exposure	2.01E+06	2.01E+07
Plant life	4.00E+01	4.00E+01
Cost of exposure	$2,000	$2,000
Cost of plant	$2,000,000,000	$2,000,000,000
Offsite damage	$2,000,000,000	$2,000,000,000
Onsite cleanup	$1,000,000,000	$1,000,000,000
Value of exposure =	$225	$22,450
Value of plant =	$0	$0
Value of offsite damage =	$106	$1,064
Value of onsite cleanup =	$0	$0
Total value of eliminating risk =	$320	$22,450