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Our ref: DCP/NRC1606

July 31, 2003

SUBJECT: Transmittal of Westinghouse Responses to Open Items Identified in the AP1000 Draft Safety Evaluation Report

This letter transmits revised Westinghouse responses to open items identified in the AP1000 Draft Safety Evaluation Report (DSER) that was issued on June 16, 2003. A list of the DSER Open Item responses that are transmitted with this letter is provided in Attachment 1. Attachment 2 provides the DSER Open Item responses.

Please contact me if you have questions regarding this transmittal.

Very truly yours,

M. M. Corletti

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

/Attachments

1. Table 1, "List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1606"

2. Westinghouse Non-Proprietary Responses to US Nuclear Regulatory Commission DSER Open Items dated July 31, 2003

D063

Attachment 1

"List of Westinghouse's Responses to DSER Open Items
as Transmitted in DCP/NRC1606"

July 31, 2003

Attachment 1

Table 1 **"List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1606"**	
3.6.3.4-1 Rev. 1	17.3.2-2 17.3.2-3 Rev. 1
5.3.3-1	17.5-1
8.2.3.1-1 Rev. 1	18.11.3.5-1 Rev. 1
13.3-1 Rev. 1	19.2.6-1 Rev. 1
14.2.10-1 Rev. 1	19A.2-1 Rev. 1
14.3.2-5 Rev. 1	19A.2-2 Rev. 1
14.3.2-7 Rev. 1	19A.2-3 Rev. 1
14.3.2-9 Rev. 1	19A.2-4 Rev. 1
14.3.2-11 Rev. 1	19A.2-5 Rev. 1
14.3.2-14 Rev. 1	19A.2-6 Rev. 1
14.3.2-15 Rev. 1	19A.2-7 Rev. 1
14.3.3-5 Rev. 1	19A.2-9 Rev. 1
14.3.3-17 Rev. 1	19A.3-1 Rev.1
	19A.3-2 Rev. 1
	19A.3-3 Rev. 1

DCP/NRC1606
Docket 52-006

July 31, 2003

Attachment 1

Westinghouse Non-Proprietary Responses to
AP1000 Draft Safety Evaluation Report (DSER) Open Items

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.6.3.4-1 (Response Revision 1)

Original RAI Number(s): 251.004

Summary of Issue:

In RAI 251.004, the staff requested that the applicant address the following: (1) clarify whether Alloy 600 material, which is susceptible to PWSCC as indicated by the V.C. Summer primary loop leakage, will be used in any of the AP1000 LBB candidate piping systems, (2) provide test and plant operational data demonstrating that the proposed weld material, Alloy 52/152, is not susceptible to PWSCC, and (3) provide an inspection plan licensees would perform to address additional inspection techniques for detecting tight flaws that might exist in LBB piping welds.

The applicant's response to RAI 251.004 states the following: (1) Alloy 600 will not be used for any of the AP1000 LBB candidate piping systems; (2) Alloy 52/152 weld material (for Alloy 690 base material) has been used in various applications such as steam generator welds and safe end-nozzle welds for 9 plants (7 years in one application) without any reported instances of environmental degradation, and although laboratory data for Alloy 52/152 in simulated primary water is limited, they indicated no environmentally-related crack propagation was observed for periods up to 4122 hours; and (3) since Alloy 52/152 weld material has better crack resistance than Alloy 82/182, augmented inservice inspection using eddy current testing (ET) to supplement ASME Code required ultrasonic testing (UT) should not be necessary for the AP1000 applications.

The staff considers the information provided for (1) to be complete and that no further information is required. Regarding (2), although the chrome content of Alloy 52/152 is approximately twice the chrome content of Alloy 82/182, making Alloy 52/152 more resistant to PWSCC, the test and plant operational data for Alloy 52/152 are for periods less than 7 years. This is not long enough for the NRC staff to consider the question of PWSCC for Alloy 52/152 material in the AP1000 LBB candidate piping to be resolved, considering the licensing period for AP1000 facilities.

To address this issue for currently operating plants, the industry has undertaken an initiative to (1) develop overall inspection and evaluation guidance, (2) assess the current inspection technology, and (3) assess the current repair and mitigation technology. An interim industry report, "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for U.S. PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," was published in April 2001 to justify the continued operation of PWRs while the industry completes the development of the final report. The final industry report on this issue has not yet been published. Subsequent to staff review and evaluation of the final report and receipt of additional Inconel UT inspection data from the industry, the staff will determine if additional regulatory actions will need to be imposed to address the potential for PWSCC to occur in lines with currently approved LBB analyses in operating plants. To address this issue for the AP1000 application, the applicant needs to modify its DCD Tier 2 Section 3.6.4 on COL information to indicate that COL holders should implement inspection plans, evaluation criteria, and other types of measures imposed on or

adopted by operating PWRs with currently approved LBB applications as part of the resolution of concerns regarding the potential for PWSCC in those units. This is Open Item 3.6.3.4-1.

Westinghouse Response Revision 1:

Based on discussions with NRC staff at a public meeting held on July 11, 2003, Westinghouse proposes to revise the COL action item to address NRC comments. DCD section 3.6.4.4 will be revised as shown.

Design Control Document (DCD) Revision:

3.6.4.4 Primary System Inspection Program for Leak-before-Break Piping

Combined License applicants referencing the AP1000 certified design will develop an inspection program for piping systems qualified for leak-before-break. The inspection program will consider the operating experience of the materials used in the AP1000 piping systems qualified for leak-before-break, and will **include augmented inspection plans and evaluation criteria consistent with those measures imposed on or adopted by operating PWRs as part of the on-going resolution of concerns regarding the potential for PWSCC in operating plants. The AP1000 inspection program will be consistent with the inspection program that is adopted for operating PWRs that employ Alloy 690, 52 and 152 in approved leak-before-break applications.** ~~consider the need for augmented inspections to those required by the applicable portions of the ASME code~~.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 5.3.3-1

Original RAI Number(s): 251.018

Summary of Issue:

The staff requested, in RAI 251.018, that the applicant demonstrate that the P-T limits are in accordance with Appendix G to 10 CFR Part 50. The applicant responded, that the AP1000 heatup and cooldown operating curves were generated using the most limiting adjusted reference temperature values and the NRC-approved methodology as documented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," with staff approved exceptions.

One exception is that instead of using best estimate fluence values, the applicant is using fluence values that are calculated fluence values. The staff finds this acceptable because this is in compliance with RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The other exception is that the KIc critical stress intensities are used in place of the KIa critical stress intensities. This methodology is taken from staff approved ASME Code Case N-641. The staff found the applicant's responses acceptable because the AP1000 P-T limit curves were developed in accordance with 10 CFR Part 50, Appendix G, with the exception that the flange requirement is in accordance with WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." Currently, the staff has not approved WCAP 15315. Any changes to the RV closure head requirements would be incorporated into Appendix G of 10 CFR Part 50. If a relaxation to 10 CFR Part 50, Appendix G is approved, this will allow the operating window to be wider. Since applicants using AP1000 are required to meet the requirements of 10 CFR Part 50, Appendix G, applicants using AP1000 must meet the closure head requirements of Appendix G of 10 CFR Part 50. However, the AP1000 DCD does not provide limitations (values of RTNDT) for the closure flange region of the RV and head. The AP1000 design must include these limitations in order to satisfy Appendix G of 10 CFR Part 50. The applicant should provide these limitations that are consistent with the present TSs and 10 CFR Part 50, Appendix G, or provide closure flange limitations with new TSs that are consistent with 10 CFR Part 50, Appendix G. This is Open Item 5.3.3-1.

Westinghouse Response:

Since it is recognized that the elimination of the flange requirement, as discussed in WCAP 15315, results in plant safety and operational improvements, Westinghouse proposes to maintain the P/T curves without the flange requirement in the AP1000 DCD and request exemption from the 10 CFR Part 50 Appendix G flange limits. Westinghouse requests further interaction with the NRC staff to resolve any technical issues associated with this exemption.

When evaluating the request for exemption, consideration should be given to the COL item in DCD Section 5.3.6.1 in which it is recognized that the P/T curves given in the DCD are generic

curves and that the Combined License Applicant is committed to addressing P/T curves based on the as-procured reactor vessel material. An LTOPS evaluation, including assessment of the RHR relief valve setpoint and relief capacity, is also committed to be performed to determine the impact of any changes in the P/T curves.

A review of the ITAAC associated with the normal RHR system relief valve (Tier 1 Section 2.3.6) shows that specification of the relief valve capacity based on the generic P/T curves in the DCD is inconsistent with the COL item in Section 5.3.6.1. The COL item requires an evaluation of the adequacy of the normal RHR system relief valve based on the P/T curves developed for the as-procured reactor vessel material, which could result in a revised required relief valve capacity. The ITAAC associated with the normal RHR system relief valve will be revised to a more general requirement so that this ITAAC is compatible with the possibility of changes in the required capacity of the valve as a result of P/T curves based on as-procured reactor vessel material.

Design Control Document (DCD) Revision:

From DCD Tier 1, Section 2.3.6, Table 2.3.6-4, page 2.3.6-12:

Table 2.3.6-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.a) The Class 1E equipment identified in Tables 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	A report exists and concludes that the Class 1E equipment identified in Table 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
7.b) The Class 1E components identified in Table 2.3.6-1 are powered from their respective Class 1E division.	Testing will be performed on the RNS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.3.6-1 when the assigned Class 1E division is provided the test signal.
7.c) Separation is provided between RNS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.

8.a) The RNS preserves containment integrity by isolation of the RNS lines penetrating the containment.	See Tier 1 Material, subsection 2.2.1, Containment System.	See Tier 1 Material, subsection 2.2.1, Containment System.
8.b) The RNS provides a flow path for long-term, post-accident makeup to the RCS.	See item 1 in this table.	See item 1 in this table.
9.a) The RNS provides LTOP for the RCS during shutdown operations.	i) Inspections will be conducted on the low temperature overpressure protection relief valve to confirm that the capacity of the vendor code plate rating is greater than or equal to system relief requirements. ii) Testing and analysis in accordance with the ASME Code Section III will be performed to determine set pressure.	i) The rated capacity recorded on the valve vendor code plate is not less than ~~650 gpm~~**the flow required to provide low-temperature overpressure protection for the RCS, as determined by the LTOPS evaluation based on the P/T curves developed for the as-procured reactor vessel material.** ii) A report exists and concludes that the relief valve opens at a pressure such that the relief capacity is not less than ~~650 gpm at a pressure of 900 psig.~~**the flow required to provide low-temperature overpressure protection for the RCS.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 8.2.3.1-1 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Because of certain electrical failures (such as a loss of isophase bus) power from the generator or grid may not be available to the RCPs for a minimum of 3 seconds following a turbine trip. The COL applicant must perform a failure modes and effects analysis (FMEA) to ensure that the design provides power to the RCPs for a minimum of 3 seconds following a turbine trip. If the power to the RCPs cannot be maintained for 3 seconds, then the DCD Tier 2 Chapter 15 analysis should be re-analyzed and provided to the staff for review. This is COL Action Item 8.2.3.1-3. Inclusion of this COL information in the DCD is Open Item 8.2.3.1-1.

Westinghouse Response:

The Chapter 15 analyses treat electrical system failures as initiating events. These initiating events are covered by the analyses described in DCD sections 15.2.6, "Loss of ac Power to the Plant Auxiliaries" and 15.3.2, "Complete Loss of Forced Reactor Coolant Flow." Note that for the first event, offsite power is assumed to be lost at the time of reactor trip. For the second event, loss of power to the RCPs occurs before the reactor trip. Therefore, for accidents initiated by electrical system failures, the RCPs are not assumed to have power following the reactor trip. Random independent failures of electric systems (such as loss of isophase bus in addition to another initiating event) are not assumed in the DCD Chapter 15 analyses. Therefore, a failure modes and effects analysis would not provide any additional value and is not required for this non-safety system. This criteria was also used for AP600.

NRC Additional Comments:

Westinghouse should justify the assumption that the failure of the isophase bus within the 3-second window following a turbine trip is not credible.

The statement in DCD Tier 2 Section 8.2.2 is confusing and should be revised to state that the requirement for RCP power following a turbine trip is assuming no electrical failures.

Westinghouse Additional Response:

The isophase bus is a passive component that must be operational for the turbine-generator to be operated. Because the isophase bus is required for power operation, it is known to be operational at the start of the 3-second time period. The failure of a passive component that is known to be initially operational within a 3-second window is an incredibly low probability event.

Therefore, it is not reasonable to assume independent failure of the isophase bus in the 3-second window following a turbine trip.

DCD Section 8.2.2 will be revised as shown below to resolve the confusion regarding the COL requirement.

Design Control Document (DCD) Revision:

Revise DCD Tier 2 Section 8.2.2 as shown:

> If, during power operation of the plant, a turbine trip occurs, the motive power (steam) to the turbine will be removed. The generator will attempt to keep the shaft rotating at synchronous speed (governed by the grid frequency) by acting like a synchronous motor. The reverse-power relay monitoring generator power will sense this condition and, after a time delay of at least 15 seconds, open the generator breaker. During this delay time the generator will be able to provide voltage support to the grid if needed. The reactor coolant pumps will receive power from the grid for at least 3 seconds following the turbine trip. The Combined License applicant will perform a grid stability analysis to show **that, with no electrical system failures**, the grid will remain stable and the reactor coolant pump bus voltage will remain above the voltage required to maintain the flow assumed in the Chapter 15 analyses for a minimum of three (3) seconds following a turbine trip. **In the Chapter 15 analyses, if the initiating event is an electrical system failure (such as failure of the isophase bus), the analyses do not assume operation of the reactor coolant pumps following the turbine trip.** The Combined License applicant will set the protective devices controlling the switchyard breakers with consideration given to preserving the plant grid connection following a turbine trip.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 13.3-1 Revision 1 Response

Original RAI Number(s): None

Summary of Issue:

<u>13.3.3.3.4 TSC as a Vital Area</u>
According to Section 2.6 of NUREG-0696, the intent of the TSC is to provide direct management and technical support to the control room during an accident. Section II.B.2 of NUREG-0737 states that any area which will or may require occupancy to permit an operator to aid in the mitigation of, or recovery from, an accident is designated as a "vital area;" and that the control room and TSC must be included among those areas where access is considered vital after an accident. Further, the design dose rate for personnel in a vital area should be such that the guidelines of GDC 19 will not be exceeded during the course of the accident. GDC 19 requires that adequate radiation protection be provided, such that dose to personnel should not be in excess of 0.05 Sv (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident. In addition, Subsection 8.2.1.f of Supplement 1 to NUREG-0737 states that the TSC will be provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 0.05 Sv (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident. These guidelines form the basic radiological habitability criteria for the TSC.

Section H.1 of NUREG-0654/FEMA-REP-1, Rev. 1, calls for establishment of a TSC in accordance with NUREG-0696. Section 2.6 of NUREG-0696 states that since the TSC is to provide direct management and technical support to the control room during an accident, it shall have the same radiological habitability as the control room under accident conditions, and the TSC ventilation system shall function in a manner comparable to the control room ventilation system. If the TSC becomes uninhabitable, the TSC plant management function shall be transferred to the control room.

As discussed above, the applicant states in DCD Tier 2 Section 18.8.3.5 that the TSC has no emergency habitability requirements, and that this is consistent with NUREG-0737. Given NUREG-0737's designation of the TSC in Section II.B.2 as a vital area, having related radiation protection criteria of GDC 19 during the course of an accident, the statement that the TSC "has no emergency habitability requirements" is not consistent with NUREG-0737. In the applicant's additional response to RAI 472.003, the apparent inconsistency is acknowledged as "confusing." The statement was removed from DCD Tier 2 Section 18.8.3.5.

Despite the removal of the statement that the TSC has no emergency habitability requirements in DCD Tier 2 Section 18.8.3.5, the design of the ventilation systems for the TSC and MCR does not provide the TSC with the same radiological habitability as the MCR under <u>all</u> accident conditions. Section 2.1 of NUREG-0696 provides that "[l]icensees who cannot meet the criteria for location, size, and habitability for the TSC must submit to NRC a request for an exception. This request must include justification for the exception and an alternate proposal. The NRC will

review requests for exceptions on a case-by-case basis." The AP1000 DCD does not request an exception to the habitability criteria for the TSC. In addition, the use of criteria different from those set forth in NUREG-0696, NUREG-0737, and Supplement 1 of NUREG-0737, will be accepted only if the substitute criteria provides a basis for determining that the applicable regulatory requirements are met.

The applicant further states in its additional response to RAI 472.003, that "[i]n practical terms, the TSC does have emergency habitability capabilities comparable to those of operating plants as long as electrical power is available either from offsite power or from the onsite diesel generators." This does not comport with the TSC emergency habitability criteria of NUREG-0696, NUREG-0737, and Supplement 1 to NUREG-0737. The staff has identified the inability of the TSC to provide emergency habitability under accident conditions as Open Item 13.3-1.a.

Westinghouse Response (to 13.3-1a):

The TSC is designed to meet GDC 19 limits during accident conditions. This is consistent with the guidance of NUREG-0696 section 2.6, Habitability, and NUREG-0737. The DCD states that the VBS meets GDC 19 under the "Abnormal Plant Operation" heading of DCD 9.4.1.2.3.1. "The main control room/technical support center HVAC equipment and ductwork that form an extension of the main control room/technical support center pressure boundary limit the overall infiltration (negative operating pressure) and exfiltration (positive operating pressure) rates to those values shown in Table 9.4.1-1. Based on these values, the system is designed to maintain operator doses within allowable General Design Criteria (GDC) 19 limits."

The AP1000 ventilation system serving the TSC exceeds the guidance of NUREG-0696 as it is redundant, instrumented in the control room and is automatically activated. NUREG-0696 section 2.6 states, "The TSC ventilation system need not be seismic Category I qualified, redundant, instrumented in the control room, or automatically activated to fulfill its role."

NUREG-0696 guidance does not suggest that the TSC meet habitability requirements all of the time. Section 2.6 of the NUREG states, "If the TSC becomes uninhabitable, the TSC plant management function shall be transferred to the control room." The existence of this statement is acknowledgment that there may be times when the TSC habitability could be challenged. This acknowledgement is logical given the fact that the ventilation system redundancy and qualification guidance of NUREG-0696 are less stringent than those for the control room ventilation system.

Based on the above, Westinghouse believes that AP1000 meets the NUREG-0696 section 2.6 guidance to "... have the same radiological habitability as the control room under accident conditions." Westinghouse also believes that it has met all applicable requirements and guidance associated with providing TSC habitability.

NRC Additional Comment:

Westinghouse should explicitly state in the DCD that when VBS is operating, it is designed to maintain the TSC within allowable GDC 19 limits for design basis accidents.

Westinghouse Additional Response:

Westinghouse will revised DCD 9.4.1.2.3.1 as identified in the "Design Control Document (DCD) Revision:" portion of this response to address the NRC comment. DCD 9.4.1.2.3.1 has also been revised to clarify that in the event of a loss of the plant ac electrical system, the VBS supplemental air filtration system can be manually transferred to the onsite standby diesel generators.

Summary of Issue (continued):

13.3.3.3.5 Isolation of MCR from TSC
DCD Tier 2 Section 18.8.3.5 further states that "[t]he TSC complies with the habitability requirements of Reference 27 [i.e., Supplement 1 to NUREG-0737] when electrical power is available." The reference to "when electrical power is available" is but one, of two, triggering events that would automatically isolate the MCR from the TSC. The second triggering event is "High-high particulate or iodine radioactivity in MCR air supply" (see DCD Section 6.4.4). In addition, the second triggering event is not reflected in DCD Tier 2 Section 3.1.2, "Protection by Multiple Fission Product Barriers," which states under Criterion 19, "Control Room," that "[i]f the normal main control room ventilation system is inoperable or if no ac power sources are available, the emergency control room habitability system automatically isolates the main control room and provides operator habitability requirements." If, for example, electrical power was available, while at the same time there was high-high particulate or iodine radioactivity in the MCR air supply, the MCR would automatically isolate from the TSC. As such, the TSC would no longer be able to ensure compliance with the radiological protection requirements of GDC 19, and therefore, the TSC would be unable to comply with the radiological habitability criteria of Supplement 1 to NUREG-0737 (i.e., Reference 27). Hence, the statement that the TSC complies with the habitability requirements of Supplement 1 to NUREG-0737 when electrical power is available, is incomplete. Addressing this concern, the applicant stated the following in their additional response to RAI 472.003.

> Should a "high-high" radiation signal or if a station blackout of more than 10 minutes occur, the VBS stops, isolates the MCR envelop and the VES begins operation to protect the MCR operators. If the system has power and is operating, it will prevent a "high-high" radiation signal. This is the reason DCD [Tier 2 Section] 18.8.3.5 states, "The TSC complies with the habitability requirements of Reference 27 [i.e., Supplement 1 to NUREG-0737] when electrical power is available."

This response is somewhat confusing. The isolation of the MCR envelop can occur with either a high-high radiation signal or loss of power. That means that isolation can occur on a high-high

radiation signal only, without loss of power. The statement that "[i]f the system has power and is operating, it will prevent a "high-high" radiation signal" implies that a high-high radiation signal will never occur, except upon loss of power. The need for the high-high radiation signal as a trigger to automatically isolate the MCR is, therefore, not needed, since the isolation already occurs upon loss of power. Subsequent high-high radioactivity would be inconsequential, as the MCR would have already been isolated from the TSC upon loss of power, with potential loss of TSC habitability. These habitability concerns should be resolved. This is identified as Open Item 13.3-1.b.

Westinghouse Response (to 13.3-1b):

As stated in the response to DSER Open Item 13.3-1.a., Westinghouse believes that AP1000 meets all applicable requirements and guidance associated with providing TSC habitability. As for VBS operation, Westinghouse provides the following discussion, which hopefully will clarify how the system, including isolation signals, is intended to function.

The only events that would shutdown VBS would be a loss of power or multiple failures to the redundant systems. These events are no different than the events that would cause the HVAC systems serving the TSC in a conventional plant to shutdown. A "high-high" radiation signal would not occur if VBS is operating properly. If VBS is operating properly, it is filtering the air, as well as providing a positive pressure in both the MCR and the TSC which precludes a "high-high" signal from being generated. In the case where there is a loss of power, VBS would isolate the MCR after a period of 10 minutes. The 10 minute delay allows for the high probability that the on-site standby diesel generators will start, thereby restoring power to the plant and to VBS. The delay also minimizes isolating the control room and actuating VES when it is not necessary. Should there be a coincident high radiation event during the loss of power event however, VBS would not delay 10 minutes, but would instead immediately isolate the main control room. Therefore, the only time that the "high-high" isolation is "needed" is in the 10 minute period following a loss of power to the VBS. It is however good engineering practice to provide diverse parameters to actuate safety systems. Thus, the statements in the DCD, which identify that isolation of the MCR envelope can occur with either a "high-high" radiation signal or loss of power and; that the TSC complies with the habitability requirements of Supplement 1 to NUREG-0737 when electrical power is available are correct and consistent with the design.

Westinghouse is not proposing specific word changes to the DCD at this time to address VBS operation. However, we are amenable to such word changes if it helps to resolve this issue.

NRC Additional Comments:

While staff recognizes Westinghouse's efforts to clarify the main control room emergency habitability system (VES) triggering events, the following further clarification is still needed. The following listed DCD sections and July 7, 2003, letter are still a bit inconsistent (and confusing). While the July 7, 2003, letter provides further clarification regarding staff's concern associated

with the triggering events for the VES, the letter's reference to "a coincident high radiation event during the loss of power event" does not seem to be reflected in the AP1000 DCD sections. Suggest these sections, and any other related sections, be revised to be consistent with one another, and reflect what appears to be an apparent THIRD triggering event for actuation of the VES. This third triggering event appears to be "a coincident high radiation event during the loss of power event (less than, or equal to, 10 minutes)."

References (in part):

(1) AP1000 DCD, Tier 2, Section 3.1.2, Criterion 19 - Control Room; p. 3.1-11 (Rev. 0)

"If the normal main control room ventilation system is inoperable or if no ac power sources are available, the emergency control room habitability system automatically isolates the main control room and provides operator habitability requirements."

(2) AP1000 DCD, Tier 2, Section 6.4; p. 6.4-1 (Rev. 0)

"When a source of ac power is not available to operate the nuclear island nonradioactive ventilation system or radioactivity is detected in the MCR air supply, which could lead to exceeding General Design Criterion 19 operator dose limits, the main control room emergency habitability system (VES) is capable of providing emergency ventilation and pressurization for the main control room."

(3) AP1000 DCD, Tier 2, Section 6.4.3.2; p. 6.4-7 (Rev. 0)

"Operation of the main control room emergency habitability system is automatically initiated by the following safety-related signals:

- High-high particulate or iodine ratioactivity [sic] in the main control room supply air duct
- Loss of ac power"

(4) AP1000 DCD, Tier 2, Section 6.4.4; p. 6.4-9 (Rev. 0)

"Automatic transfer of habitability system functions from the nuclear island nonradioactive ventilation system to the main control room emergency habitability system is accomplished by the receipt of one of two signals:

- "High-high" particulate or iodine radioactivity in MCR air supply
- Loss of ac power sources"

(5) AP1000 DCD, Tier 2, Section 9.4.1.2.3.1; p. 9.4-11 (Rev. 0)

"If ac power is unavailable for more than ten (10) minutes or if "high-high" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding GDC 19 operator dose limits, the plant safety and monitoring system automatically isolates the main control room from the normal main control room/technical support center HVAC subsystem by closing the supply, return, and toilet exhaust isolation valves. Main control room habitability is maintained by the main control room emergency habitability system which is discussed in Section 6.4."

(6) Westinghouse July 7, 2003, letter, response to 13.3-1b

"In the case where there is a loss of power, VBS would isolate the MCR after a period of 10 minutes. The 10 minute delay allows for the high probability that the on-site standby diesel generators will start, thereby restoring power to the plant and to VBS. The delay also minimizes isolating the control room and actuating VES when it is not necessary. Should there be a coincident high radiation event during the loss of power event however, VBS would not delay 10 minutes, but would instead immediately isolate the main control room. Therefore, the only time that the "high-high" isolation is "needed" is in the 10 minute period following a loss of power to the VBS."

(7) AP1000 DCD, Tier 1, Section 2.7.1; p. 2.7.1-1 (Rev. 0)

"In addition, the VBS isolates the HVAC penetrations in the main control room boundary on high-high particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system (VES)."

Westinghouse Additional Response:

Westinghouse will revised the DCD as identified in the "Design Control Document (DCD) Revision:" portion of this response to improve the consistency of the description of the VES triggering events. Please note that there is no "third" triggering event leading to the actuation of VES. The "high radiation event" referred to in our earlier response to the DSER open item and contained in the phrase "a coincident high radiation event during the loss of power event" is not meant to describe actuation logic, but rather a generic condition in which high radiation exists.

Design Control Document (DCD) Revision:

Tier 2, 1.9, Issue 83 - Control Room Habitability; revise the 1st and 2nd paragraphs under AP1000 Response as follows:

Habitability of the main control room ~~during normal operation~~ is provided by the **main control room/technical support center HVAC subsystem of the** nonsafety- related nuclear island

nonradioactive ventilation system (VBS). **If ac power is unavailable for more than ten (10) minutes or if "high-high" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES).** ~~In the event of a design basis accident involving a radiation release or a loss of all ac power event, the nonsafety-related nuclear island nonradioactive ventilation system is automatically terminated, the main control room pressure boundary is isolated, and the safety-related main control room emergency habitability system (VES) is actuated.~~ The safety-related main control room emergency habitability system supplies breathable quality air for the main control room operators while the main control room is isolated.

In the event of external smoke or radiation release, the nonsafety-related nuclear island nonradioactive ventilation system provides for a supplemental filtration mode of operation, as discussed in DCD Section 9.4. ~~In the event of a Hi-Hi radiation level, the safety-related main control room emergency habitability system is actuated~~. In the unlikely event of a toxic chemical release, the safety-related main control room emergency habitability system has the capability to be manually actuated by the operators. Further, a 6-hour supply of self-contained portable breathing equipment is stored inside the main control room pressure boundary.

Tier 2, 3.1.2, Criterion 19 - Control Room; revise the 3rd and 4th paragraphs under AP1000 Compliance as follows:

The main control room is shielded by the containment and auxiliary building from direct gamma radiation and inhalation doses resulting from the postulated release of fission products inside containment. Refer to Chapter 15 for additional information on accident conditions. **The main control room/technical support center HVAC subsystem of the nuclear islandnonradioactive ventilation system (VBS)**~~The normal main control room ventilation system is provided to~~ **allows** access to and occupancy of the main control room under accident conditions as described in subsection 9.4.1. Sufficient shielding and the~~normal~~ **main control room/technical support center HVAC subsystem**~~ventilation system~~ provide adequate protection so that personnel will not receive radiation exposure in excess of 5 rem whole-body or its equivalent to any part of the body for the duration of the accident.

If ac power is unavailable for more than ten (10) minutes or if "high-high" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits,~~If the normal main control room ventilation system is inoperable or if no ac power sources are available~~,the **protection and safety monitoring system**~~emergency control room habitability system~~ automatically isolates the main control room and ~~provides~~ operator habitability requirements **are thenmet by the main control room emergency habitability system (VES)**. The ~~emergency~~ main control room **emergency** habitability system also ~~provides~~ **allows** access to and occupancy of the main control room under accident conditions. The emergency main control room habitability system is designed to satisfy seismic Category I requirements as described in Section 3.2 and the system design is described in Section 6.4.

Tier 2, 6.4; revise the 3rd paragraph as follows:

If ac power is unavailable for more than ten (10) minutes or if "high-high" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES)~~. When a source of ac power is not available to operate the nuclear island nonradioactive ventilation system or radioactivity is detected in the MCR air supply, which could lead to exceeding General Design Criterion 19 operator dose limits, the~~ **The** main control room emergency habitability system ~~(VES)~~ is capable of providing emergency ventilation and pressurization for the main control room.

Tier 2, 6.4.3.2; revise the 1st paragraph as follows:

Operation of the main control room emergency habitability system is automatically initiated by ~~the~~ **either of the** following ~~safety-related signals~~**conditions**:

- "High-high" particulate or iodine ~~ratioactivity~~ **radioactivity** in the main control room supply air duct
- Loss of ac power **for more than ten (10) minutes**

Tier 2, 6.4.4; revise the 3rd from last paragraph as follows:

Automatic transfer of habitability system functions from the **main control room/technical support center HVAC subsystem of the** nuclear island nonradioactive ventilation system to the main control room emergency habitability system is **initiated by either of the following conditions**~~: accomplished by the receipt of one of two signals~~:

- "High-high" particulate or iodine radioactivity in MCR air supply **duct**
- Loss of ac power **for more than ten (10) minutes**~~sources~~

Tier 2, 9.4.1.2.3.1; revise the last sentence of the 2nd paragraph under Abnormal Plant Operation as follows: (Note: The second to last sentence is also shown below. It has no changes but is included for contextual purposes only.)

The main control room/technical support center HVAC equipment and ductwork that form an extension of the main control room/technical support center pressure boundary limit the overall infiltration (negative operating pressure) and exfiltration (positive operating pressure) rates to those values shown in Table 9.4.1-1. Based on these values, the system is designed to maintain ~~operator~~ **personnel** doses within allowable General Design Criteria (GDC) 19 limits **during design basis accidents in both the main control room and the technical support center.**

Tier 2, 9.4.1.2.3.1; revise the last sentence of the 3rd paragraph under Abnormal Plant Operation as follows:

If ac power is unavailable for more than ten (10) minutes or if "high-high" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding GDC 19 operator dose limits, the ~~plant~~ **protection and** safety ~~and~~ monitoring system automatically isolates the main control room from the normal main control room/technical support center HVAC subsystem by closing the supply, return, and toilet exhaust isolation valves. Main control room habitability is maintained by the main control room emergency habitability system which is discussed in Section 6.4.

Tier 2, 9.4.1.2.3.1; revise the last sentence of the 3rd to last paragraph under Abnormal Plant Operation as follows:

Power is supplied to the main control room/technical support center HVAC subsystem by the plant ac electrical system. In the event of a loss of the plant ac electrical system, the main control room/technical support center ventilation subsystem ~~is automatically~~ **can be** transferred to the onsite standby diesel generators.

Tier 1, 2.7.1; revise the last sentence of 1st paragraph under Design Description as follows:

In addition, the VBS isolates the HVAC penetrations in the main control room boundary on "high-high" particulate or iodine **radioactivity** ~~concentrations~~ in the main control room supply air **duct** or on ~~extended~~ a loss of ac power **for more than ten (10) minutes.** ~~to~~ **This action** supports operation of the main control room emergency habitability system (VES).

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.2.10-1 (Response Revision 1)

Original RAI Number(s): 261.009, 261.016

Summary of Issue:

RG 1.68, Appendix A, Item 4.c recommends performance of pseudo-rod ejection testing to verify calculation models and accident analysis assumptions during low power testing. The NRC staff could not locate an AP1000 low power test abstract that describes this testing. In RAI 261.009, the NRC staff requested that the applicant provide additional information regarding the performance of pseudo-rod ejection testing for the AP1000 design. In their November 13, 2002, RAI response, the applicant stated that sufficient test data has been obtained from previous plant startups and that licensees of new plants need only to confirm calculational models. The applicant also provided several licensing precedents associated with this position.

The NRC staff lacked sufficient information to accept the applicant's position regarding performance of low power psuedo-rod ejection testing. As described in the staff evaluation of RAI 261.007b, Item 2, below, the NRC staff requested that the applicant provide additional information relating to the conduct of pseudo-rod ejection testing. This request for additional information is identified as RAI 261.016. Pending resolution of RAI 261.016 and RAI 261.009, this is Open Item 14.2.10-1.

Westinghouse Response:

The responses to RAI 261.009 Rev. 0 and RAI 261.016 Rev. 0 were transmitted to the NRC via DCP/NRC1532 dated 11/15/02 and DCP/NRC1588 dated 05/13/03, respectively.

NRC Additional Comments:

Westinghouse should provide more of a basis for why the pseudo-rod-ejection test is performed at the 30 to 50% power range.

Westinghouse Additional Response:

As stated previously, Westinghouse performs the pseudo-rod-ejection test in the 30% to 50% power range. The test is performed on the first unit only as part of the rod cluster control assembly out of bank measurements test. This 30 to 50% range is the preferred range in which to perform the test because the range is sufficiently low so as not to cause the plant to exceed peaking factor limits, yet sufficiently high so as to validate calculation tools and accident analysis assumptions. While it may be possible to perform the test at higher power levels, there is no advantage in doing so. Testing at higher power levels has the undesirable effect of increasing the risk of exceeding peaking factor limits while not providing any "better" validation of the calculation tools and accident analysis assumptions. Performing the pseudo-rod-ejection test in the 30 to 50% power range is also consistent with testing that was performed on current generating Westinghouse plants. WCAP 7905 Revision 1 provides analysis that supports the

performance of this test at the 30% to 50% power range. Once the calculation tools and accident analysis assumptions are confirmed to be consistent with design expectations, it is not necessary to perform the test on additional units.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-5 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Section 2.3.5, "Mechanical Handling System," the design description (items 3.b and 3.c) for the equipment hatch hoist and the maintenance hatch hoist are not identified as single failure proof as they are in Tier 2. In addition to not being identified as single failure proof, Table 2.3.5.2 does not require a test, inspection, or analysis to demonstrate whether these items of equipment will meet their design criteria. As such, the design description in Tier 2 is inconsistent with that of the ITAAC. This is Open Item 14.3.2-5.

NRC Follow-On Comment:

Westinghouse has revised DCD Section 9.1.5.3 to make the maintenance hatch hoist non-single failure proof, but still says the maintenance hatch hoist is operational after a seismic event.

Westinghouse Response (Revision 1):

Based on a review of the AP600 heavy load analyses for the equipment and maintenance hatch hoists, Westinghouse has revised the classification of the AP1000 maintenance hatch hoist to a non-single failure proof design which is consistent with the AP600 classification. **Although, the maintenance hatch is not single failure proof, the Seismic I classification for the hoist is maintained to ensure the ability to close the hatch after a seismic event.** Coincident with this change Westinghouse has revised the associated ITAAC to delete the design commitment for the maintenance hatch hoist and to provide a design commitment for the equipment hatch hoist related to the single failure nature of the design. These changes make the DCD Tier 1 and Tier 2 information on the maintenance and equipment hatch hoists consistent.

These changes were incorporated into AP1000 DCD Revision 5, which was transmitted to the NRC via Westinghouse letter DCP/NRC1593 Dated May 19, 2003. The changes incorporated into DCD Revision 5 are given below.

Design Control Document (DCD) Revision:

The following changes **were made as part of the original Westinghouse response and have** been incorporated into DCD Revision 5.

From Tier 1, pages 2.3.5-1 through 2.3.5-4:

2.3.5 Mechanical Handling System

Design Description

The mechanical handling system (MHS) provides for lifting heavy loads. The MHS equipment can be operated during shutdown and refueling.

The component locations of the MHS are as shown in Table 2.3.5-3.

1. The functional arrangement of the MHS is as described in the Design Description of this Section 2.3.5.
2. The seismic Category I equipment identified in Table 2.3.5-1 can withstand seismic design basis loads without loss of safety function.
3. The MHS provides the following safety-related functions:
   a) The containment polar crane prevents the uncontrolled lowering of a heavy load.
   b) The equipment hatch hoist **prevents the uncontrolled lowering of a heavy load.**~~positions the hatch to minimize loss of water inventory from containment during loss of shutdown cooling events.~~
   ~~c)The maintenance hatch hoist positions the hatch to minimize loss of water inventory from containment during loss of shutdown cooling events.~~
4. The spent fuel shipping cask crane cannot move over the spent fuel pool.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.5-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the MHS.

Table 2.3.5-1

Equipment Name	Tag No.	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety Function
Containment Polar Crane	MHS-MH-01	Yes	No/No	Avoid uncontrolled lowering of heavy load.
Equipment Hatch Hoist	MHS-MH-05	Yes	No/No	~~Positions hatch to minimize loss of water inventory from containment during loss of shutdown cooling events.~~**Avoid uncontrolled lowering of heavy load.**
~~Maintenance Hatch Hoist~~	~~MHS-MH-06~~	~~Yes~~	~~No/No~~	~~Positions hatch to minimize loss of water inventory from containment during loss of shutdown cooling events.~~

Table 2.3.5-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the MHS is as described in the Design Description of this Section 2.3.5.	Inspection of the as-built system will be performed.	The as-built MHS conforms with the functional arrangement as described in the Design Description of this Section 2.3.5.
2. The seismic Category I equipment identified in Table 2.3.5-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.5-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.5-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
3.a) The containment polar crane prevents the uncontrolled lowering of a heavy load.	Load testing of the main and auxiliary hoists that handle heavy loads will be performed. The test load will be at least equal to the weight of the reactor vessel head and integrated head package.	The crane lifts the test load, and lowers, stops, and holds the test load with the hoist holding brakes.
3.b) The equipment hatch hoist **prevents the uncontrolled lowering of a heavy load.** ~~positions the hatch to minimize loss of water inventory from containment during loss of shutdown cooling events.~~	~~Testing of the equipment hatch hoist will be performed.~~ **Testing of the redundant hoist holding mechanisms for the equipment hatch hoist that handles heavy loads will be performed by lowering the hatch at the maximum operating speed.**	~~The equipment hatch hoist will operate as required to move the hatch to the closed position.~~ **Each hoist holding mechanism stops and holds the hatch.**
~~3.c) The maintenance hatch hoist positions the hatch to minimize loss of water inventory from containment during loss of shutdown cooling events.~~	~~Testing of the maintenance hatch hoist will be performed.~~	~~The maintenance hatch hoist will operate as required to move the hatch to the closed position.~~

4. The spent fuel shipping cask crane cannot move over the spent fuel pool.	Testing of the spent fuel shipping cask crane is performed.	The spent fuel shipping cask crane does not move over the spent fuel pool.

Table 2.3.5-3		
Component Name	**Tag No.**	**Component Location**
Containment Polar Crane	MHS-MH-01	Containment
Equipment Hatch Hoist	MHS-MH-05	Containment
~~Maintenance Hatch Hoist~~	~~MHS-MH-06~~	~~Containment~~
Spent Fuel Shipping Cask Crane	MHS-MH-02	Auxiliary Building

From DCD page 3.2-27, Table 3.2-3:

Table 3.2-3 (Sheet 8 of 67)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Main Turbine and Generator Lube Oil System (LOS)					Location: Turbine Building
System components are Class E					
Mechanical Handling System (MHS)					Location: Various
MHS-MH-01	Containment Polar Crane	C	I	ASME NOG-1	
MHS-MH-05	Equipment Hatch Hoist	C	I	Manufacturer Std.	
MHS-MH-06	Maintenance Hatch Hoist	CD	I	Manufacturer Std.	
Balance of system components are Class E					
Main Steam System (MSS)					Location: Turbine Building
System components are Class E					
Main Turbine System (MTS)					Location: Turbine Building
System components are Class E					

From DCD page 9.1-37:

9.1.5.1 Design Basis

9.1.5.1.1 Safety Design Basis

Section 3.2 identifies safety and seismic classifications for mechanical handling system equipment. Heavy load handling systems are generally classified as nonsafety-related, nonseismic systems. The components of single-failure-proof systems necessary to prevent uncontrolled lowering of a critical load are classified as safety-related. The polar crane and the equipment hatch ~~and maintenance hatch~~ hoists are single-failure-proof systems and are classified as seismic Category I. They are designed to support a critical load during and after a safe shutdown earthquake. The equipment and maintenance hatches are required to be operational after the event.

From DCD page 9.1-38:

9.1.5.2 System Description

Table 9.1-5 lists heavy load handling systems in the nuclear island. The polar crane is designed according to the requirements of ASME NOG-1 for a Type I, single-failure-proof crane. A description of the polar crane is provided in this subsection. The equipment ~~and maintenance~~ hatch hoist systems incorporates single-failure-proof features based on NUREG-0612 guidelines. Based on the conservative design of these heavy load handling systems and associated special lifting devices, slings and load lift points (See subsection 9.1.5.2.3), a load drop of the critical loads handled by the polar crane or the equipment hatch hoist is unlikely. Except for the containment polar crane and the equipment ~~and maintenance~~ hatch hoists, the heavy load handling systems are not single-failure-proof.

From DCD page 9.1-41:

9.1.5.3 Safety Evaluation

The design and arrangement of heavy load handling systems promotes the safe handling of heavy loads by one of the following means:

- A single-failure-proof system is provided so that a load drop is unlikely.
- The arrangement of the system in relationship to safety-related plant components is such that the consequences of a load drop are acceptable per NUREG 0612.Postulated load drops are evaluated in the heavy loads analysis.

The polar crane and the equipment ~~and maintenance~~ hatch hoist systems are single failure proof. These systems stop and hold a critical load following the credible failure of a single component. Redundancy is provided for load bearing components such as the hoisting ropes, sheaves, equalizer assembly, hooks, and holding brakes. These systems are designed to support a critical load during and after a safe shutdown earthquake. The equipment and maintenance hatch hoist systems are designed to remain operational following the event. The polar crane is designed

to withstand rapid pressurization of the containment during a design basis loss of coolant accident or main steam line break, without collapsing.

The spent fuel shipping cask storage pit is separated from the spent fuel pool. The spent fuel shipping cask crane cannot move over the spent fuel pool because the crane rails do not extend over the pool. Mechanical stops prevent the spent fuel shipping cask crane from going beyond the ends of the rails.

A heavy loads analysis is performed to evaluate postulated load drops from heavy load handling systems located in safety-related areas of the plant, specifically the nuclear island. No evaluations are required for critical loads handled by the containment polar crane or the equipment ~~and maintenance~~ hatch hoists, since a load drop is unlikely. |

The heavy loads analysis is to confirm that a postulated load drop does not cause unacceptable damage to reactor fuel elements, or loss of safe shutdown or decay heat removal capability.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **14.3.2-7 (Response Revision 1)**

Original RAI Number(s): **420.048d**

Summary of Issue:

Section 2.3.19, "Communication Systems." ITAACs have not been identified for the communication system (EFS) as discussed in Tier 2 Section 9.5.2 beyond those given in Tables 2.3.19-2, and 3.1-1 (Emergency Response Facilities). There is no assurance that the appropriate tests and confirmatory criteria will be accomplished to meet regulatory requirements, especially 10 CFR 73.55(e)-(g) and noise level considerations for worse case postulated noise levels. The applicant needs to provide appropriate ITAAC for all the communication systems. This is Open Item 14.3.2-7.

Westinghouse Response:

A response to this issue was provided in response to RAI 420.048 (item d) transmitted by Westinghouse letter DCP/NRC1590, dated May 14, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

The term "emergency response facility communications" is confusing and should be revised to indicate that this system is for offsite communications.

DCD Tier 1 Section 3.1 Design Description items 2 and 4 do not match Tier 2. Westinghouse should resolve these discrepancies.

Westinghouse Additional Response:

DCD Tier 2 will be revised as shown below.

No changes were made regarding Section 3.1 item 4. After reviewing DCD Tier 2 Section 18.8.3.6, there does not appear to be a discrepancy between Tier 1 and Tier 2 for this item.

Design Control Document (DCD) Revision: (Revised Response)

Table 1.8-2 (Sheet 4 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
9.1-6	Radiation Monitor	9.1.6
9.3-1	Air Systems (NUREG-0933 Issue 43)	9.3.7
9.4-1	Ventilation Systems Operations	9.4.12
9.5-1	Qualification Requirements for Fire Protection Program	9.5.1.8
9.5-2	Fire Protection Analysis Information	9.5.1.8
9.5-3	Regulatory Conformance	9.5.1.8
9.5-4	NFPA Exceptions	9.5.1.8
9.5-5	Operator Actions Minimizing Spurious ADS Actuation	9.5.1.8
9.5-6	Offsite Interfaces	9.5.2.5.1
9.5-7	Emergency ~~Response Facility~~**Offsite** Communications	9.5.2.5.2

9.5.2 Communication System

The communication system (EFS) provides effective intraplant communications and effective plant-to-offsite communications during normal, maintenance, transient, fire, and accident conditions, including loss of offsite power. The communication system consists of the following subsystems:

- Wireless telephone system
- Telephone/page system
- Private automatic branch exchange (PABX) system
- Sound-powered system
- Emergency ~~response facility~~**offsite** communications
- Security communication system.

9.5.2.5.2 Emergency ~~Response Facility~~Offsite Communications

The emergency ~~response facility~~**offsite** communication system, including the crisis management radio system, will be addressed by the Combined License applicant.

14.2.9.4.13 Plant Communications System Testing

Purpose

The purpose of the plant communications system testing is to verify that the as-installed components properly perform the functions of verifying the proper operation and adequacy of the plant communication systems used during normal and abnormal operations, as described in Section 9.5.

Prerequisites

The construction testing of the communication system has been completed. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance Criteria

Plant communications system performance is observed and recorded during a series of individual component and integrated system testing. The inplant communications system includes the following subsystems:

- Wireless telephone system
- Telephone/page system
- Private Automatic Branch Exchange (PABX) System
- Sound Powered Phone System
- Emergency ~~Response Facility~~**Offsite** Communication System
- Security Communication System

18.8.3.5 Technical Support Center Mission and Major Tasks

The mission of the technical support center (TSC) is to provide an area and resources for use by personnel providing plant management and technical support to the plant operating staff during emergency evolutions. The TSC relieves the reactor operators of peripheral duties and communications not directly related to reactor system manipulations and prevents congestion in the control room.

Communications needs are established for the staff within the TSC, and between the TSC and the plant **(including the main control room and operational support center), the emergency operations facility**, the Combined License holder management, outside authorities **(including the NRC)** and the public.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-9 (Response Revision 1)

Original RAI Number(s): 252.010

Summary of Issue:

In RAI 252.001, the staff requested information related to the geometry, fabrication, materials, accessibility for inspection, and operating conditions for control rod drive system penetrations, as motivated by recent operating experience. See NRC Bulletins 2001-01, 2002-01 and 2002-02. Since the RAI was issued, the staff has issued Orders, EA-03-009, to operating license holders related to inspection for cracks in these penetrations and attachment welds. The staff subsequently issued followup questions to the applicant related to changes in design and fabrication to reduce residual stresses, the ability to visually inspect 360 degrees around each nozzle, preservice volumetric inspection, and determination of operating head temperature. The applicant responded to the followup questions in a letter dated April 7, 2003. Please provide proposed ITAAC related to the issues noted above and which were discussed in your RAI responses. This is Open Item 14.3.2-9.

NRC Follow-On Comment:

Provide an ITAAC on top-of-the head visual inspection, including 360 degrees around each of the reactor vessel head penetration nozzles.

Westinghouse Response (Revision1):

~~Westinghouse provided the response to this Open Item in the response to RAI 252.010, which was transmitted to the NRC via Westinghouse letter DCP/NRC1592 (dated May 21, 2003)~~.

Westinghouse will revise Tier 1 Section 2.1.3 (Reactor System ITAAC) to include a design commitment to perform a top-of-the head visual inspection, including 360 degrees around each of the reactor vessel head penetration nozzles.

Design Control Document (DCD) Revision:

~~None~~

From DCD Tier 1, Revision 6, Section 2.1.3 Reactor System, Design Description, page 2.1.3-2:

11. The reactor pressure vessel (RPV) beltline material has a Charpy upper-shelf energy of no less than 75 ft-lb.

12. Safety-related displays of the parameters identified in Table 2.1.3-1 can be retrieved in the main control room (MCR).

13. The fuel assemblies and rod control cluster assemblies intended for initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.

14. **A top-of-the head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle, can be performed.**

From DCD Tier 1, Section 2.1.3, Reactor System, Table 2.1.3-2, page 2.1.3-8:

Table 2.1.3-2 (cont.)
Inspections, Tests, Analysis, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analysis	Acceptance Criteria
11. The RPV beltline material has a Charpy upper-shelf energy of no less than 75 ft-lb.	Testing of the Charpy V-Notch specimen of the RPV beltline material will be performed.	A report exists and concludes that the initial RPV beltline Charpy upper-shelf energy is no less than 75 ft-lb.
12. Safety-related displays of the parameters identified in Table 2.1.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.1.3-1 can be retrieved in the MCR.

13. The fuel assemblies and rod control cluster assemblies intended for initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.	An analysis is performed of the reactor core design.	A report exists and concludes that the fuel assemblies and rod cluster control rod assemblies intended for the initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.

14. A top-of-the head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle, can be performed.	**A preservice visual examination of the reactor vessel head top surface and penetration nozzles will be performed.**	**A report exists that documents the results of the top-of-the head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-11 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

The staff reviewed Tier 2 Section 5.3.4 as it applies to pressurized thermal shock in accordance with SRP 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock." Section 50.61 of 10 CFR Part 50, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," defines the fracture toughness requirements for protection against pressurized thermal shock (PTS) events. The requirements in 10 CFR 50.61 establish the PTS screening criteria, below which no additional action is required for protection from PTS events. The screening criteria are given in terms of reference temperature (R_{TPTS}). These criteria are 148.0°C (300°F) for circumferential welds and 132.2°C (270°F) for plates, forgings, and axial welds. To verify that the design will be in accordance with the regulatory requirements associated with PTS, the applicant needs to provide an appropriate ITAAC. The following is a suggested design commitment for this ITAAC: The amount of copper and nickel in the reactor vessel materials and the projected neutron fluences for the 40 year period of the COL will result in R_{TPTS} values lower than the screening criteria contained in 10 CFR 50.61. This is Open Item 14.3.2-11.

Original Westinghouse Response:

In DCD section 5.3.3.1 Westinghouse commits to the use of reactor vessel material in which the nickel and copper content are limited to values less than those given in DCD Table 5.3-1. The AP1000 generic pressure-temperature curves are developed considering a radiation embrittlement of up to 54 effective full power years (60 year design life with 90 percent availability). These are generic, limiting curves for the AP1000 based on the reactor vessel material maximum copper and nickel content as given in DCD Table 5.3-1. The resulting end-of-life RT_{PTS} values are committed to be less than the screening criteria given in 10 CFR 50.61. DCD Table 5.3-3 provides preliminary RT_{PTS} values for the AP1000 of 66 F and 98 F for the reactor vessel beltline forging and beltline weld, respectively. These values are well below the screening criteria as shown in DCD Table 5.3-3.

There are also Combined License applicant commitments provided in DCD section 5.3.6 to address verification of plant-specific belt line material properties and to develop plant specific pressure-temperature curves based on the copper and nickel content of the actual material.

The reactor vessel design commitments and Combined License applicant commitments in the DCD are sufficient to ensure the design will be in accordance with the regulatory requirements associated with PTS without the addition of a new ITAAC. This is consistent with other recently certified new plant designs, including AP600 and System 80+, that do not include an ITAAC related to regulatory requirements associated with PTS.

NRC Follow-On Comment:

Propose a COL item to do a PTS evaluation based on as-procured vessel material data.

Westinghouse Response to NRC Follow-On Comment:

The Combined Operating License item in DCD Section 5.3.6.4 related to reactor vessel material will be revised to include a pressurized thermal shock evaluation based on as-procured vessel material data.

Design Control Document (DCD) Revision:

~~None~~
From DCD Revision 6, Section 5.3.6, page 5.3-23:

5.3.6.4 Reactor Vessel Materials Properties Verification

The Combined License applicant will address verification of plant-specific belt line material properties consistent with the requirements in subsection 5.3.3.1 and Tables 5.3-1 and 5.3-3. **The verification will include a pressurized thermal shock evaluation based on as procured reactor vessel material data. This evaluation report will be submitted for NRC staff review.**

The verification will include structural analysis of the AP1000 reactor vessel insulation and support structure.

PRA Revision:

None

DSER Open Item Number: 14.3.2-14 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Section 3.3, ITAAC Table 3.3-6, Acceptance Criteria 2.g states that the tolerance on the height of the containment vessel is +12", -6" and the tolerance on the inside diameter is also +12", -6". The information included in Tier 2 related to the containment design does not address the +12" tolerance on the inside diameter. All of the applicant's analyses, calculations, and responses to the RAIs related to the containment vessel are based on the nominal inside diameter of 130 feet. From its review, it is the staff's understanding that the vessel wall inside diameter, currently specified for 130'-0", marginally meets ASME Code allowable. Adding 1 foot to the vessel diameter will reduce the design margin. The applicant should justify the use of the proposed tolerances. This is Open Item 14.3.2-14.

Westinghouse Response (Revision 1):

The tolerances given in Section 3.3, ITAAC Table 3.3-6, Acceptance Criteria 2.g relate only to its function as a heat transfer surface and as the boundary of the containment volume (see Design Commitment 2.g). The tolerances in Acceptance Criteria 2.g do not apply to its function as a pressure vessel. The containment vessel is covered as a component of the Containment System in Section 2.2.1, ITAAC Table 2.2.1-3, Acceptance Criteria 2.a which requires the existence of the ASME Code Section III design report for the as-built containment vessel. In addition, the key design characteristics of the containment vessel are designated as Tier 2* (see response to DSER Open Item 14.3.2-3).

The ASME Code Section III, Division 1, Subsection NE requires that: "For components subjected to internal pressure, the inside diameter shall be taken as the nominal inner face . . ." It goes on to state that: "The difference between the maximum and minimum inside diameters [of the fabricated vessel] at any cross section shall not exceed 1% of the nominal diameter at the cross section under consideration." It then requires a report be prepared as an addendum to the design report that compares the final as-built vessel to its design report. Differences must be justified or the design report must be revised. As a result, if the as-built inner diameter deviates from the design inner diameter, the difference must be addressed in the as-built reconciliation. No changes are required to the current ITAAC table.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

DSER Open Item Number: 14.3.2-15 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Section 3.7, "Design Reliability Assurance Program" (D-RAP). The staff found that the list of risk significant components in Table 3.7-1 was not updated to include all risk-significant structures, systems, and components (SSCs) from the list of risk significant SSCs identified in Tier 2 Section 17.4, Table 17.4-1, "Risk Significant SSCs within the Scope of D-RAP." Specifically, the list of risk significant components should include:

- Compressed and Instrument Air System Air Compressor Transmitter
- Passive Containment Cooling System Diverse (3rd) Motor Operated Drain Isolation Valve function
- In-containment Refueling Water Storage Tank Vents
- Normal Residual Heat Removal Valve V055 function
- Feedwater Isolation Valves

As discussed in Section 17.4 of this report, the staff determined that Table 17.4-1 contained an acceptable list of risk significant SSCs under the scope of D-RAP. In Table 17.4-1, the applicant also removed the safety related passive core cooling condensate sump re-circulation valves' automatic open function from the D-RAP for the AP1000 design and this should be reflected in ITAAC Table 3.7-1. This is Open Item 14.3.2-15.

Westinghouse Response:

We have performed a review of the DCD D-RAP Table 17.4-1 and ITAAC Table 3.7-1. Based on that review we have the following comments:

1. The PRA importance of the Compressed and Instrument Air System, Air Compressor Pressure Transmitter has been re-evaluated. Based on the current AP1000 PRA this instrument just meets the DRAP selection criteria (RAW, RRW) for LRF although it does not meet the DRAP selection criteria for CDF. Furthermore, it has been determined that **the PRA models are conservative which resulted in the RAW / RRW values for this instrument being over estimated. The conservatism is due to not modeling air bottles that are provided for the SFW control valves. If these air bottles had been modeled in the PRA, the failure of the pressure instrument in the CAS would have a reduced PRA importance. As a result, it should no longer be listed in the DRAP tables in the DCD or the ITAAC. Therefore it has been removed from DCD Table 17.4-1 and it has not been added to ITAAC Table 3.7-1. Note that the SFW control valve air bottles are not risk important because if they are not available, the CAS can provide air to the SFW control valves.**

2. We agree that that the following items should be added to ITAAC Table 3.7-1:
   - IRWST vents
   - Main Feedwater Isolation Valves

3. The 3rd PCS water drain valve does not have to be added to ITAAC Table 3.7-1 because it is already listed in the table. Under item PCCWST Drain Isolation Valves are listed 3 valves, PCS-PL-V001A/B/C. The C valve is the diverse (3rd) drain valve.

4. We agree that RNS valve 055 should be added. However, as indicated in DCD Table 17.4-1, other RNS MOVs are also required to allow the RNS to provide RCS makeup following ADS actuation, including:
   - V011 RNS discharge containment isolation
   - V022 RNS suction containment isolation
   - V055 RNS suction from the SFS Cask Loading Pit
   - V062 RNS suction from the IRWST

5. We agree that the PXS containment recirculation MOVs (PXS-PL-V117A/B) should be removed from ITAAC Table 3.7-1, since they have been removed from DCD Table 17.4-1.

6. **We have revised ITAAC Table 3.7-1 to list the components alphabetically by system as is done in DCD Table 17.4-1. In addition, we have added tag numbers to DCD Table 17.4-1. Note that the tag numbers shown in both tables are a simplified format (PXS-PL-V014A/B instead of PXS-PL-V014A, PXS-PL-V014B). The names of the components were made consistent in both tables. In the process of revising these tables, we found that a few additional changes are required to make the tables consistent.**

   Additional changes to DCD Table 17.4-1.
   - Add CVS makeup pump suction and discharge check valves.
   - Add inverters and battery chargers for the 24 hour batteries
   - Add reactor vessel insulation water inlet and steam vent devices
   - Add reactor cavity doorway damper
   - Add service water cooling tower fans
   - Add **air cooled chillers and pumps**
   - Add **onsite** diesel generator room cooling fans
   - Add fuel assemblies
   - **Add Note 5 to list the containment isolation valves controlled by DAS.**
   - **Add Note 6 to list PLS controls included in this DRAP item.**
   - **Add references to DCD Tables 7.2-2 and 7.3-1 for lists of the reactor trips and ESF actuations under the PMS software and hardware.**
   - Remove PXS valves PXS-PL-V125A/B from the IRWST injection squib valve group since these valves are not squibs and –V123A/B and –V125A/B lists the four squibs in these lines.
   - **Remove the turbine impulse pressure sensors (001 and 002) from the DAS. These sensors do not provide input to DAS in the AP1000.**

- Remove the PLS logic cabinet for CVS functions since this is redundant to the PLS actuation hardware item.

Additional changes to ITAAC Table 3.7-1:

- Add main feedwater flow and startup feedwater flow sensors
- Remove PXS-PL-V124A/B from the IRWST injection squib valves and add to the IRWST injection check valves.
- Change PLS item to be "control functions" instead of "automatic control functions" because some of these controls are only manual controls. This change affects the title of ITAAC Table 3.7-2.

Design Control Document (DCD) Revision:

DCD Table 17.4-1 will be changed to add tag numbers as is shown in ITAAC Table 3.7-1 and to include the changes indicated above, as shown below:

DCD Table 17.4-1 (Sheet 1 of 10)

RISK-SIGNIFICANT SSCS WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
~~System: Compressed and Instrument Air System (CAS)~~		
~~Air Compressor Transmitter~~	~~RRW/LRF~~	~~Failure of air compressor transmitter.~~
System: Component Cooling Water (CCS)		
~~CCS~~ Component Cooling Water Pumps (CCS-MP-01A/B)	EP	These pumps provide cooling of the normal residual heat removal system (RNS) and the spent fuel pool heat exchanger. Cooling the RNS heat exchanger is important to investment protection during shutdown reduced-inventory conditions. CCS valve realignment is not required for reduced-inventory conditions.
System: Containment System (CNS)		
Containment Vessel (CNS-MV-01)	EP, L2	The containment vessel provides a barrier to steam and radioactivity released to the atmosphere following accidents.
Hydrogen Igniters (VLS-EH-1 through -60)	EP, L2, Regulations	The hydrogen igniters provide a means to control H_2 concentration in the containment atmosphere, consistent with the hydrogen control requirements of 10 CFR 50.34f.
System: Chemical and Volume Control System (CVS)		
~~CVS~~ Makeup Pumps (CVS-MP-01A/B)	RAW/CCF	These pumps provide makeup to the RCS to accommodate leaks and to provide negative reactivity for shutdowns, steam line breaks, and ATWS.
~~CVS~~ Makeup Pump Suction and Discharge Check Valves (CVS-PL-V113, -V160A/B)	RAW	These CVS check valves are normally closed and have to open to allow makeup pump operation.
System: Diverse Actuation System (DAS)		

Table 17.4-1 (Sheet 2 of 10)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
DAS Processor Cabinets and Control Panel ~~Actuation Hardware~~ (used to provide automatic and manual actuation~~sensor input through control output and indication~~) **(DAS-JD-001, -002, OCS-JC-020)**	RAW	The DAS is diverse from the PMS and provides automatic and manual actuation of selected plant features including control rod insertion, turbine trip, passive residual heat removal (PRHR) heat exchanger actuation, core makeup tank actuation, isolation of critical containment lines, and passive containment cooling system (PCS) actuation.
Annex Building UPS Distribution Panels (EDS1-EA-14,~~-and~~ EDS2-EA-14)	RAW	These panels distribute power to the DAS equipment.
~~Control~~ Rod Drive MG Sets (Field Breakers) (PLS-MG-01A/B)	RAW	These breakers open on a DAS reactor trip signal demand to de-energize the control rod MG sets and allow the rods to drop.
Containment Isolation Valves Controlled by DAS (Note 5)	RAW	These containment isolation valves are important in limiting offsite releases following core melt accidents.
~~Turbine Impulse Pressure Transmitters 001 and 002~~	~~RAW~~	~~These sensors provide signals used as permissives for the DAS automatic reactor trip function.~~
System: Main AC Power System (ECS)		
Reactor Coolant Pump Switchgear~~Circuit Breakers~~ (ECS-ES-31, -32, -41, -42, -51, -52, -53, -54)	RAW/CCF	These breakers open automatically to allow core makeup tank operation.
Ancillary Diesel Generators (ECS-MS-01, -02)	EP	For post-72 hour actions, these generators are available to provide power for Class 1E monitoring, MCR lighting and for refilling the PCS water storage tank and spent fuel pool.
System: Main and Startup Feedwater System (FWS)		
Startup Feedwater Pumps (FWS-MP-03A/B	EP	The startup feedwater system pumps provide feedwater to the steam generator. This capability provides an alternate core cooling mechanism to the PRHR heat exchangers for non-loss-of-coolant-accidents or steam generator tube ruptures.
System: General I&C[4]		

Low Pressure/DP Sensors - IRWST level sensors (PXS-045, -046, -047, -048)	RAW/CCF	The in-containment refueling water storage tank (IRWST) level sensors support PMS ~~and DAS~~ functions. They are utilized in automatic actuation and they provide indications to the operator. IRWST level supports IRWST recirculation actions.

Table 17.4-1 (Sheet 3 of 10)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
High Pressure/DP Sensors - **RCS Hot Leg Level (RCS-160A/B)** - **Pressurizer Pressure (RCS-191A/B/C/D)** - **Pressurizer Level (RCS-195A/B/C/D)** - SG Narrow-Range Level **(SGS-001, -002, -003, -004, -005, -006, -007, -008)** - **SG Wide-Range Level (SGS-011, -012, -013, -014, -015, -016, -017, -018)** - **Main Steamline** Pressure **(SGS-030/-031/-032/-033/-034/-035/-036/-037)** - **Main Feedwater Wide-**Range Flow (SGS-050A/C/E, -051A/C/E) - Startup Feedwater Flow (SGS-055A/B, -056A/B)	RAW/CCF	The following sensors are included in this group. These sensors support PMS~~, DAS~~ and PLS functions. They are utilized in reactor trip and ESF functions, and provide indications to the operator. Main feedwater flow sensors support startup feedwater actuation and startup feedwater flow sensors support PRHR actuation. The hot leg level sensors automatically actuate the IRWST **injection** and ~~provide information to the operator for manual actuation of the~~ automatic depressurization system (ADS) **valves during shutdown conditions.**
CMT Level Sensors (PXS-011A/B/C/D, -012A/B/C/D, -013A/B/C/D, -014A/B/C/D)	RAW/CCF	These level sensors provide input for automatic actuation of the ADS. They also provide indications to the operator.
System: Class 1E DC Power and Uninterruptible Power System (IDS)		

125 Vdc 24-hour Batteries, Inverters, and Chargers **(IDSA-DB-1A/B, IDSB-DB-1A/B, IDSC-DB-1A/B, IDSD-DB-1A/B, IDSA-DU-1, IDSB-DU-1, IDSC-DU-1, IDSD-DU-1, IDSA-DC-1, IDSB-DC-1, IDSC-DC-1, IDSD-DC-1)**	RAW/CCF	The batteries provide power for the PMS and safety-related valves. The chargers are the preferred source of power for Class 1E dc loads and are the source of charging for the batteries. The inverters provide uninterruptible ac power to the I&C system.

Table 17.4-1 (Sheet 4 of 10)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
System: Passive Containment Cooling System (PCS)		
125 Vdc Distribution Panels **(IDSA-DD-1, -EA-1/2, IDSB-DD-1, -EA-1/2/3, IDSC-DD-1, -EA-1/2/3, IDSD-DD-1, -EA-1/2)**	RAW	These panels distribute power to components in the plant that require 1E dc power support.
Fused Transfer Switch Boxes **(IDSA-DF-1, IDSB-DF-1, IDSC-DF-1, IDSD-DF-1)**	RAW	The fused disconnect switches connect the different levels of Class 1E distribution panels.
125 Vac Motor Control Centers **(IDSA-DK-1, IDSB-DK-1, IDSC-DK-1, IDSD-DK-1)**	EP	These buses provide power for the PMS and safety-related valve operation.
~~PCCWST~~ Recirculation Pumps **(PCS-MP-01A/B)**	EP	These pumps provide the motive force to refill the PCS water storage tank during post-72 hour support actions.
PCCWST ~~Air-Operated Drain Isolation Valves and Diverse (3rd) Motor Operated~~ Drain Isolation Valves **(PCS-PL-V001A/B/C)**	EP, L2	These valves **(two AOVs and one MOV)** open automatically to drain water from a water storage tank onto the outside surface of the containment shell. This water provides evaporative cooling of the containment shell following accidents.
System: Plant Control System (PLS)		
PLS Actuation Hardware **(Control functions listed in note 6)**	**RAW/CCF**	**This common cause failure event is assumed to disable all logic outputs from the PLS associated with CVS reactor makeup, RNS reactor injection, spent fuel cooling, component cooling of RNS SFS heat exchangers, service water cooling of CCS heat exchangers, standby diesel generators, and hydrogen igniters.**
~~PLS Logic Cabinet Supporting CVS Functions~~	~~RAW/CCF~~	~~This is the distributed controller that supports the CVS function.~~

Table 17.4-1 (Sheet 5 of 10)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
System: Protection and Safety Monitoring System (PMS)		
PMS Actuation Software	RAW/CCF	**The PMS software provides the automatic reactor trip and ESF actuation functions listed in Tables 7.2-2 and 7.3-1.** ~~modules include field input signal processing, control board signal input processing, actuation logic algorithms and output logic functions.~~
PMS Actuation Hardware	RAW/CCF	**The PMS hardware provides the automatic reactor trip and ESF actuation functions listed in Tables 7.2-2 and 7.3-1.** ~~includes the following:~~ ~~Plant Protection Subsystems~~ ~~ESF Coincidence Logic~~ ~~ESF Actuation Subsystem~~ ~~Manual Input Multiplexers~~
Main Control Room (MCR) 1E Displays and System Level Controls ~~Mechanisms (to Support Operator Actions)~~ **(OCS-JC-010, -011)**	RAW/CCF	This includes the Class 1E PMS (QDPS) ~~and DAS~~ displays and controls. ~~It also includes the PLS displays and controls associated with CVS reactor makeup, RNS reactor injection, spent fuel cooling, component cooling of RNS and SFS heat exchangers, service water cooling of CCS heat exchangers, standby diesel generators, and hydrogen igniters.~~ These displays and system level controls ~~mechanisms~~ provide important plant indications ~~and variables~~ to allow the operator to monitor and control the plant during ~~normal conditions and during design basis~~ accidents.
Reactor Trip Switch~~G~~gear **(PMS-JP-RTS A01/02, B01/02, C01/02, D01/02)**	RAW/CCF	These breakers open automatically to allow insertion of the control rods.
System: Passive Core Cooling System (PXS)		
IRWST Vents **(PXS-MT-03)**	RAW/CCF	The IRWST vents provide a pathway to vent steam from the tank into the containment. The IRWST vents also have a severe accident function to prevent the formation of standing hydrogen flames close to the containment walls. This function is accomplished by designing the vents located further from the containment walls to open with less IRWST internal pressure than the other vents.

Table 17.4-1 (Sheet 6 of 10)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
IRWST Screens **(PXS-MY-Y01A/B)**	RAW/CCF	The IRWST injection lines provide long-term core cooling following a LOCA. These screens are located inside the IRWST and prevent large particles from being injected into the RCS. They are designed so that they will not become obstructed.
Containment Recirculation Screens **(PXS-MY-Y02A/B)**	RAW/CCF	The containment recirculation lines provide long-term core cooling following a LOCA. The screens are located in the containment and prevent large particles from being injected into the RCS. They are designed so that they will not become obstructed.
CMT Discharge Isolation Valves **(PXS-PL-V014A/B, PXS-PL-V015A/B)**	RAW/CCF	These air-operated valves automatically open to allow core makeup tank injection.
CMT Discharge Check Valves **(PXS-PL-V016A/B, PXS-PL-V017A/B)**		These check valves are normally open. They close during rapid accumulator injection.
Accumulator Discharge Check Valves **(PXS-PL-V028A/B, -V029A/B)**	RAW/CCF	These check valves open when the RCS pressure drops below the accumulator pressure to allow accumulator injection.
PRHR Heat Exchanger Control Valves **(PXS-PL-V108A/B)**	RAW/CCF	The PRHR heat exchangers provide core cooling following non-LOCAs, steam generator tube ruptures, and anticipated transients without scram. The air-operated valves automatically open to initiate PRHR heat exchanger operation.
Containment Recirculation Squib Valves **(PXS-PL-V118A/B, PXS-PL-V120A/B)**	RAW/CCF	The containment recirculation lines provide long-term core cooling following a LOCA. These squib valves open automatically to allow containment recirculation when the IRWST level is reduced to about the same level as the containment level. These squib valves can also allow long-term core cooling to be provided by the RNS pumps. These squib valves can provide a rapid flooding of the containment to support in-vessel retention during a severe accident.

Table 17.4-1 (Sheet 7 of 10)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
IRWST Injection Check Valves (PXS-PL-V122A/B, -V124A/B)	RAW/CCF	The containment recirculation lines provide long-term core cooling following a LOCA. These check valves open when the IRWST level is reduced to approximately the same level as the containment level.
IRWST Injection Squib Valves (PXS-PL-V123A/B, -V125A/B)	RAW/CCF	The IRWST injection lines provide long-term core cooling following a LOCA. These squib valves open automatically to allow injection when the RCS pressure is reduced to below the IRWST injection head.
IRWST Gutter Bypass Isolation Valves (PXS-PL-V130A/B)	RAW/CCF	These valves direct water collected in the IRWST gutter to the IRWST. This capability extends PRHR heat exchanger operation.
System: Reactor Coolant System (RCS)		
ADS Stages 1/2/3 ~~Motor-Operated~~ Valves (MOV) (RCS-PL-V001A/B, -V002A/B, -V003A/B, -V011A/B, -V012A/B, -V013A/B)	RAW	The ADS provides a controlled depressurization of the RCS following LOCAs to allow core cooling from the accumulator, IRWST injection, and containment recirculation. The ADS provides "bleed" capability for feed/bleed cooling of the core. The ADS also provides depressurization of the RCS to prevent a high-pressure core melt sequence.
ADS ~~4th~~ Stage 4 ~~Squib~~ Valves (Squib) (RCS-PL-V004A/B/C/D)	RAW/CCF	The ADS provides a controlled depressurization of the RCS following LOCAs to allow core cooling from the accumulator, IRWST injection, and containment recirculation. The ADS provides "bleed" capability for feed/bleed cooling of the core. The ADS also provides depressurization of the RCS to prevent a high-pressure core melt sequence.
Pressurizer Safety Valves (RCS-PL-V005A/B)	EP	These valves provide overpressure protection of the RCS.

Table 17.4-1 (Sheet 8 of 10)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
Reactor Vessel Insulation Water Inlet and Steam Vent Devices (RCS-MN-01)	EP	These devices provide an engineered flow path to promote in-vessel retention of the core in a severe accident.
Reactor Cavity Doorway Damper	EP	This device provides a flow path to promote in-vessel retention of the core in a severe accident.
Fuel Assemblies (RXS-FA-A04 through -N10)	SMA	The nuclear fuel assembly includes the fuel pellets, fuel cladding, and associated support structures. This equipment, which provides a first barrier for release of radioactivity and allows for effective core cooling, had the least margin in the seismic margin analysis.
System: Normal Residual Heat Removal System (RNS)		
Residual Heat Removal Pumps (RNS-MP-01A/B)	RAW	These pumps provide shutdown cooling of the RCS. They also provide an alternate RCS lower pressure injection capability following actuation of the ADS. The operation of these pumps is important to investment protection during shutdown reduced-inventory conditions. RNS valve realignment is not required for reduced-inventory conditions. ~~RNS valve V055 is included.~~
RNS Motor-Operated Valves (RNS-PL-V011, -V022, -V055, -V062)	RRW/FVW	These MOVs align a flowpath for nonsafety-related makeup to the RCS following ADS operation, initially from the cask loading pit and later from the containment.
System: Spent Fuel Cooling System (SFS)		
Spent Fuel Cooling~~FS~~ Pumps (SFS-MP-01A/B)	EP	These pumps provide flow to the heat exchangers for removal of the design basis heat load.

Table 17.4-1 (Sheet 9 of 10)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
System: Steam Generator System (SGS)		
Main Steam Safety Valves (SGS-PL-V030A/B, -V031A/B, -V032A/B, -V033A/B, -V034A/B, -V035A/B)	EP	The steam generator main steam safety valves provide overpressure protection of the steam generator. They also provide core cooling by venting steam from the steam generator.
Main Steam and Feedwater Isolation Valves (SGS-PL-V040A/B, -V057A/B)	RAW	The steam generator main steam and feedwater isolation valves provide isolation of the steam generator following secondary line breaks and steam generator tube rupture.
System: Service Water System (SWS)		
Service Water Pumps and Cooling Tower Fans (SWS-MP-01A/B, SWS-MA-01A/B)	EP	These pumps and fans provide cooling of the CCS heat exchanger which is important to investment protection during shutdown reduced-inventory conditions. Service water system valve realignment is not required for reduced-inventory conditions.
System: Nuclear Island Nonradioactive Ventilation System (VBS)		
VBS MCR and I&C Rooms B/C Ancillary Fans (VBS-MA-10A/B, -11, -12)	EP	For post-72 hour actions, these fans are available to provide cooling of the MCR and the two I&C rooms (B/C) that provide post-accident monitoring.
System: Chilled Water System (VWS)		
~~VWS Low Capacity Subsystem~~ Air Cooled Chillers and Pumps (VWS-MS-02, -03, VWS-MP-02, -03)	RAW/CCF	This VWS subsystem provides chilled cooling water to the CVS makeup pump room. The ~~motor-driven pumps,~~and chillers ~~and unit cooler fans~~ are important components of the VWS.
System: Onsite Standby Power System (ZOS)		

Table 17.4-1 (Sheet 10 of 10)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
~~Nonsafety-related Standby~~ Onsite Diesel Generators (ZOS-MS-05A/B)	EP	These diesels generators provide ac power to support operation of nonsafety-related equipment such as the startup feedwater pumps, CVS pumps, RNS pumps, CCS pumps, SWS pumps, and the PLS. Providing ac power to the RNS and the equipment necessary to support its operation is important to investment protection during reduced inventory conditions.
Engine ~~Standby Diesels~~ Room Air Handling Unit Supply ~~Cooling~~ Fans (VZS-MA-03A/B)	EP	These fans provide cooling of the rooms containing the onsite~~standby~~ diesel generators.

Notes:

1. Only includes equipment at the component level. Other parts of the SSC or support systems are not included unless specifically listed.
2. Definition of Rationale Terms:

   CCF = Common Cause Failure (for the SSCs whose inclusion rationale is RAW/CCF, the RAW is based on common cause failure of two or more of the specified SSCs.

   EP = Expert Panel

   RAW = Risk Achievement Worth

   RRW = Risk Reduction Worth

   SMA = Seismic Margin Analysis
3. Maintenance/surveillance recommendations for equipments are documented in each appropriate DCD section.
4. This category captures instrumentation and control equipment common cause failures across systems.
5. **The following containment isolation valves are controlled by DAS:**

   Containment Purge Inlet Containment Isolation Valve ORC VFS-PL-V003

   Containment Purge Inlet Containment Isolation Valve IRC VFS-PL-V004

   Containment Purge Discharge Containment Isolation Valve IRC VFS-PL-V009

   Containment Purge Discharge Containment Isolation Valve ORC VFS-PL-V010

   Sump Discharge Containment Isolation Valve IRC WLS-PL-V055

   Sump Discharge Containment Isolation Valve ORC WLS-PL-V057

6. **The PLS provides control of the following functions:**

CVS Reactor Makeup

RNS Reactor Injection from Cask Loading Pit

Startup Feedwater fron CST

Spent Fuel Cooling

Component Cooling of RNS and SFS Heat Exchangers

Service Water Cooling of the CCS Heat Exchangers

On-Site Diesel Generators

Hydrogen Igniters

ITAAC Table 3.7-1 will be changed as shown below:

Table 3.7-1
Risk-Significant Components

Equipment Name	Tag No.
Component Cooling Water System (CCS)	
Component Cooling Water~~S~~ Pumps	CCS-MP-01A/B
Containment System (CNS)	
Containment Vessel	CNS-MV-01
Hydrogen Igniters	VLS-EH-1 through -60
Chemical and Volume Control System (CVS)	
Makeup Pumps	CVS-MP-01A/B
Makeup Pump Suction and Discharge Check Valves	**CVS-PL-V113** **CVS-PL-V160A/B**
Diverse Actuation System (DAS)	
DAS **Processor Cabinets and Control Panel** ~~Actuation Hardware~~ (used to provide automatic and manual actuation)	DAS-JD-001 DAS-JD-002 OCS-JC-020
Annex Building UPS Distribution Panels (provide power to DAS)	EDS1-EA-14 EDS2-EA-14
~~Control~~ Rod **Drive MG Sets** (Field Breakers)	PLS-MG-01A/B
Containment Isolation Valves Controlled by DAS	Refer to Table 2.2.1-1
Main AC Power System (ECS)	
Reactor Coolant Pump Switchgear ~~Circuit Breakers~~	ECS-ES-31, -32, -41, -42 -51, -52, -61, -62
Ancillary Diesel Generators	ECS-MG-01, -02
Main and Startup Feedwater System (FWS)	
Startup Feedwater~~S~~ Pumps	FWS-MP-03A/B
General I&C	
IRWST Level Sensors	PXS-045, -046, -047, -048
RCS Hot Leg Level Sensors	RCS-160A/B
Pressurizer Pressure Sensors	RCS-191A/B/C/D

Note: Dash (-) indicates not applicable.

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Pressurizer Level Sensors	RCS-195A/B/C/D
Steam Generator Narrow-Range Level Sensors	SGS-001, -002, -003, -004, -005, -006, -007, -008
Steam Generator Wide-Range Level Sensors	SGS-011, -012, -013, -014, -015, -016, -017, -018
Main Steam Line Pressure Sensors	SGS-030, -031, -032, -033 -034, -035, -036, -037
Main Feedwater Wide-Range Flow Sensors	**SGS-050A/C/E, -051A/C/E**
Startup Feedwater Flow Sensors	**SGS-055A/B, -056A/B**
CMT Level Sensors	PXS-011A/B/C/D, -012A/B/C/D, -013A/B/C/D, -014A/B/C/D
Class 1E DC Power and Uninterruptible Power System (IDS)	
125 Vdc 24-Hour Batteries	IDSA-DB-1A/B, IDSB-DB-1A/B, IDSC-DB-1A/B, IDSD-DB-1A/B
125 Vdc 24-Hour Battery Chargers	IDSA-DC-1, IDSB-DC-1, IDSC-DC-1, IDSD-DC-1
125Vdc Distribution Panels	IDSA-DD-1, IDSA-EA-1/-2, IDSB-DD-1, IDSB-EA-1/-2/-3, IDSC-DD-1, IDSC-EA-1/-2/-3, IDSD-DD-1, IDSD-EA-1/-2
Fused Transfer Switch Boxes	IDSA-DF-1, IDSB-DF-1/-2 IDSC-DF-1/-2, IDSD-DF-1
125 Vdc Motor Control Centers	IDSA-DK-1, IDSB-DK-1, IDSC-DK-1, IDSD-DK-1
125 Vdc 24-Hour Inverters	IDSA-DU-1, IDSB-DU-1, IDSC-DU-1, IDSD-DU-1
Passive Containment Cooling System (PCS)	
~~PCCWST~~ Recirculation Pumps	PCS-MP-01A/B
PCCWST Drain Isolation Valves	PCS-PL-V001A/B/C
Plant Control System (PLS)	
PLS Actuation Software and Hardware (used to provide ~~automatic~~ control functions ~~listed in Table 3.7-2~~)	**Refer to Table 3.7-2**
Protection and Monitoring System (PMS)	
PMS Actuation Software (used to provide automatic control functions ~~listed in Tables 2.5.2-2 and 2.5.2-3~~)	**Refer to Tables 2.5.2-2 and 2.5.2-3**

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
PMS Actuation Hardware (used to provide automatic control functions ~~listed in Tables 2.5.2-2 and 2.5.2-3~~)	**Refer to Tables 2.5.2-2 and 2.5.2-3**
MCR 1E Displays and System Level Controls	OCS-JC-010, -011
Reactor Trip Switchgear	PMS-JP-RTS A01/02, B01/02, C01/02, D01/02
Passive Core Cooling System (PXS)	
IRWST Vents	**PXS-MT-03**
IRWST Screens	PXS-MY-Y01A/B
Containment Recirculation Screens	PXS-MY-Y02A/B
CMT Discharge Isolation Valves	PXS-PL-V014A/B, -V015A/B
CMT Discharge Check Valves	PXS-PL-V016A/B, -V017A/B
Accumulator Discharge Check Valves	PXS-PL-V028A/B, -V029A/B
PRHR HX Control Valves	PXS-PL-V108A/B
~~Containment Recirculation Isolation Motor-operated Valves~~	~~PXS-PL-V117A~~ ~~PXS-PL-V117B~~
Containment Recirculation Squib Valves	PXS-PL-V118A/B, -V120A/B
IRWST Injection Check Valves	PXS-PL-V122A/B, **-V124A/B**
IRWST Injection Squib Valves	PXS-PL-V123A/B, -V125A/B ~~PXS-PL-V124A~~ ~~PXS-PL-V124B~~
IRWST Gutter Bypass Isolation Valves	PXS-PL-V130A/B
Reactor Coolant System (RCS)	
ADS Stage 1/2/3 Valves (MOVs)	RCS-PL-V001A/B, -V011A/B RCS-PL-V002A/B, -V012A/B RCS-PL-V003A/B, -V013A/B
ADS Stage 4 Valves (Squibs)	RCS-PL-V004A/B/C/D
Pressurizer Safety Valves	RCS-PL-V005A/B
Reactor Vessel Insulation Water Inlet and Steam Vent Devices	**RCS-MN-01**
Reactor Cavity Doorway Damper	**-**
Fuel Assemblies	**RXS-FA-A04 through -N10**
Normal Residual Heat Removal System (RNS)	
Residual Heat Removal~~NS~~ Pumps	RNS-MP-01A/B

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
RNS Motor Operated Valves	RNS-PL-V011, -V022, -V055, -V062
Spent Fuel Cooling System (SFS)	
Spent Fuel Cooling~~S~~ Pumps	SFS-MP-01A/B
Steam Generator System (SGS)	
Main Steam Safety Valves	SGS-PL-V030A/B, -V031A/B, -V032A/B -V033A/B, -V034A/B, -V035A/B
Main Steam Line Isolation Valves	SGS-PL-V040A/B
Main Feedwater Isolation Valves	SGS-PL-V057A/B
Service Water System (SWS)	
Service Water Cooling Tower Fans	MA-01A/B
Service Water Pumps	SWS-MP-01A/B
Nuclear Island Nonradioactive Ventilation System (VBS)	
MCR Ancillary Fans	VBS-MA-10A, -10B
I&C Room B/C Ancillary Fans	VBS-MA-11, -12
Chilled Water System (VWS)	
Air Cooled Chiller Pumps	VWS-MP-02, -03
Air Cooled Chillers	VWS-MS-02, -03
Onsite Standby Power System (ZOS)	
Engine Room Air Handling Unit Supply Fans	VZS-MA-03A/B
Onsite~~Standby~~ Diesel Generators	ZOS-MS~~G~~-05~~2~~A/B

ITAAC Table 3.7-2 will be changed as shown below:

Table 3.7-2 PLS D-RAP ~~Automatic~~ Control Functions
CVS Reactor Makeup
RNS Reactor Injection from cask loading pit
Startup Feedwater from CST
Spent Fuel Cooling
Component Cooling of RNS and SFS Heat Exchangers
Service Water Cooling of CCS Heat Exchangers
Standby Diesel Generators
Hydrogen Ignitors

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **14.3.3-5 (Response Revision 1)**

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

Summary of Issue:

Section 2.5.2, Table 2.5.2-4, "PMS Manually Actuated Functions," is not consistent with the information provided in Tier 2 Table 7.2-4, "System-Level Manual Inputs to the Reactor Trip Functions," and Table 7.3-3, "System-Level Manual Inputs to the ESFAS." Tier 1 design description Item 6(c) should be modified to clarify that the functions listed on Table 2.5.2-4 are based on minimum inventory requirements. This is Open Item 14.3.3-5.

Westinghouse Response:

The functions listed in DCD Tier 1 Table 2.5.2-4 are not based on minimum inventory requirements. The minimum inventory requirements are stated in DCD Tier 1 Table 2.5.2-5. The functions listed in Table 2.5.2-4 are intended to be the PMS manual initiation functions of PMS (not limited to the minimum inventory).

Based on a review of DCD Tier 2 Tables 7.2-4 and 7.3-3, two actuation functions should be added to Table 2.5.2-4, "Chemical and Volume Control System Isolation" and "Normal Residual Heat Removal System Isolation." The other entries in Tables 7.2-4 and 7.3-3 are various block and permissive controls that are not initiation functions, do not meet the screening criteria to be included in Tier 1, and need not be added to Table 2.5.2-4.

Chemical and Volume Control System Isolation and Normal Residual Heat Removal System Isolation will be added to DCD Tier 1 Table 2.5.2-4 as shown below.

NRC Additional Comments:

The inconsistent naming between DCD Tier 1 Table 2.5.2-4 and Tier 2 Tables 7.2-4 and 7.3-3 makes it difficult to compare Tier 1 with Tier 2. Westinghouse should revise these tables for consistency.

Westinghouse should also explain how the screening criteria were applied for the manual actuations.

Westinghouse Additional Response:

The DCD revisions shown below resolve the inconsistencies between the Tier 1 and Tier 2 tables.

The Tier 1 selection (i.e., screening) criteria are described in DCD Tier 2 Section 14.3.2.1. Based on the criteria that "only the information from the Tier 2 Material that is most important to safety" are included in certified design descriptions, manual actuation of safety functions is included as a top-level function of PMS. The specific manual actuation functions are included in Table 2.5.2-4. The PMS block and interlock functions are also important, but somewhat less important than the manual actuation of the safety functions. Therefore, the automatic features of these blocks and interlocks are included in Tier 1 Design Description item 9 and Tables 2.5.2-6 and 2.5.2-7, but without specifying the details of the operator (manual) interface.

We also note that the Reactor Trip Reset listed in Tier 2 Table 7.2-4 is not treated in Tier 1. This is because the important aspect of the reactor trip function is how the reactor is tripped, not how it is reset. Therefore, Tier 1 requires the reactor to be tripped automatically as specified in Table 2.5.2-2 and manually as specified in Table 2.5.2-4, but does not specify how the reset operates.

Westinghouse believes that this is a proper application of the principle that the most important functions and features are treated with the highest level of detail and thus Tier 1 includes only a subset of Tier 2.

Design Control Document (DCD) Revision:

Table 2.5.2-4 **PMS Manually Actuated Functions**
Reactor Trip Safeguards Actuation Containment Isolation **Depressurization System Stages 1, 2, and 3** ~~ADS~~ Actuation **Depressurization System Stage 4** ~~ADS~~ Actuation ~~Main~~ Feedwater Isolation ~~CMT~~**Core Makeup Tank** Injection Actuation Steam Line Isolation Passive Containment Cooling Actuation ~~PRHR~~**Passive Residual Heat Removal** Heat Exchanger Alignment IRWST Injection ~~IRWST~~ Containment Recirculation Actuation~~/IRWST Drain to Containment~~ ~~MCR~~**Control Room** Isolation and Air Supply Initiation Steam Generator Relief Isolation **Chemical And Volume Control System Isolation** **Normal Residual Heat Removal System Isolation**

Table 2.5.2-6 PMS Blocks
Reactor Trip Functions: Source Range High Neutron Flux Reactor Trip Intermediate Range High Neutron Flux Reactor Trip Power Range High Neutron Flux (Low Setpoint) Trip Reactor Coolant Pump High Bearing Water Temperature Trip Pressurizer Low Pressure Trip Pressurizer High Water Level Trip Low Reactor Coolant Flow Trip Low Reactor Coolant Pump Speed Trip High Steam Generator Water Level Trip Engineered Safety Features: Containment Isolation Main Feedwater Isolation Reactor Coolant Pump Trip ~~CMT~~ **Core Makeup Tank** Injection Turbine Trip Steam Line Isolation Startup Feedwater Isolation Block of Boron Dilution ~~CVS Makeup Line~~**Chemical and Volume Control System** Isolation Steam Dump Block Auxiliary Spray and Letdown Purification Line Isolation ~~PRHR~~ **Passive Residual Heat Removal** Heat Exchanger Alignment Normal Residual Heat Removal System Isolation

Table 7.2-2 (Sheet 1 of 2)

REACTOR TRIPS

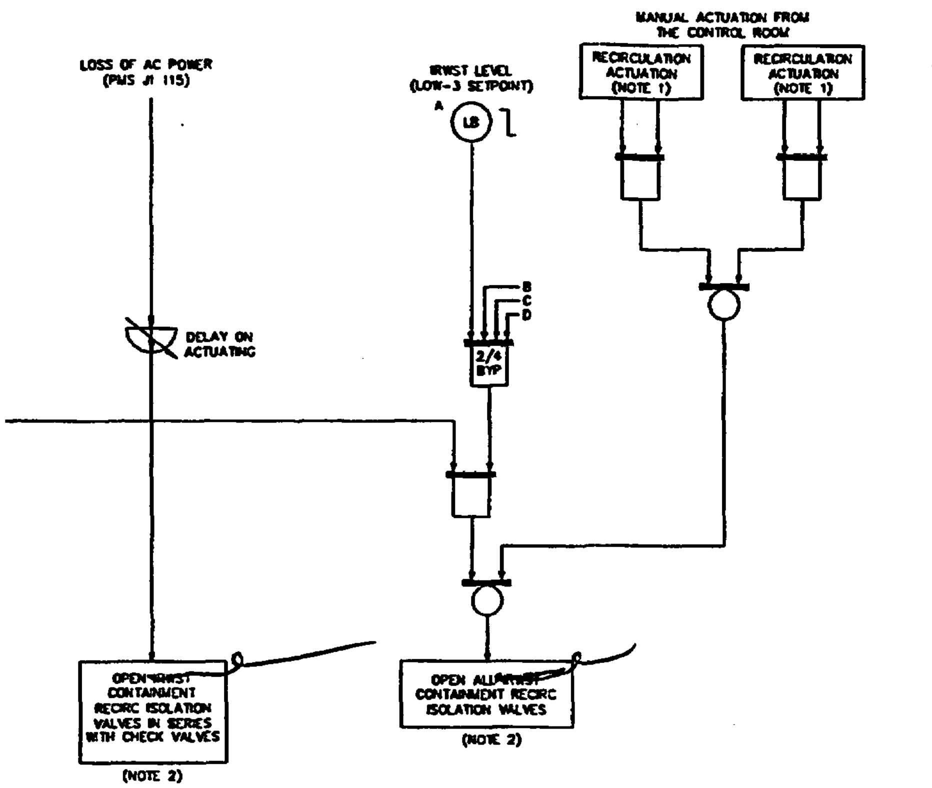
Reactor Trip(1)	No. of Channels	Division Trip Logic	Bypass Logic	Permissives and Interlocks (See Table 7.2-3)
Source Range **High Neutron Flux** Reactor Trip	4	2/4	Yes(2)	P-6, P-10
Intermediate Range **High Neutron Flux** Reactor **Trip**	4	2/4	Yes(2)	P-10
Power Range **High Neutron Flux** (Low Setpoint) Trip	4	2/4	Yes(2)	P-10
Power Range **High Neutron Flux**	4	2/4	Yes(2)	----

(High Setpoint) Trip

Table 7.2-4

SYSTEM-LEVEL MANUAL INPUTS TO THE REACTOR TRIP FUNCTIONS

Manual Control	To Divisions				Figure 7.2-1 Sheet
Manual Reactor Trip Control #1	A	B	C	D	2 & 13
Manual Reactor Trip Control #2	A	B	C	D	2 & 13
Reactor Trip Reset	A	B	C	D	13
Source Range High Neutron Flux Block, Division A	A				3
Source Range High Neutron Flux Block, Division B		B			3
Source Range High Neutron Flux Block, Division C			C		3
Source Range High Neutron Flux Block, Division D				D	3
Intermediate Range High Neutron Flux Block, Division A	A				3
Intermediate Range High Neutron Flux Block, Division B		B			3
Intermediate Range High Neutron Flux Block, Division C			C		3
Intermediate Range High Neutron Flux Block, Division D				D	3
Power Range High Neutron Flux Block (Low Setpoint), Division A	A				3
Power Range High Neutron Flux Block (Low Setpoint), Division B		B			3
Power Range High Neutron Flux Block (Low Setpoint), Division C			C		3
Power Range High Neutron Flux Block (Low Setpoint), Division D				D	3
Manual Safeguards Actuation Control #1	A	B	C	D	2 & 11
Manual Safeguards Actuation Control #2	A	B	C	D	2 & 11
Manual Core Makeup Tank Injection Control #1	A	B	C	D	2 & 12
Manual Core Makeup Tank Injection Control #2	A	B	C	D	2 & 12
Manual Depressurization System Stages 1, 2, & 3 Actuation Controls #1 & 2	A	B	C	D	2 & 15
Manual Depressurization System Stages 1, 2, & 3 Actuation Controls #3 & 4	A	B	C	D	2 & 15



ION CONSISTS OF FOUR MOMENTARY CONTROLS.
CONTROLS ARE OPERATED SIMULTANEOUSLY.
JUR IN ALL DIVISIONS.

L INDIVIDUALLY SEALED IN (LATCHED), SO
ACTUATION SIGNAL WILL NOT CAUSE THESE
TURN TO THE CONDITION HELD PRIOR TO THE
UATION SIGNAL.

CONSISTS OF SEPARATE MOMENTARY CONTROLS.
NTROL WILL RESET A SEPARATE DIVISION.

ION ALSO OCCURS DURING CONTAINMENT
J1 113.

Figure 7.2-1 (Sheet 16 of 20)

Functional Diagram
In-Containment Refueling Water Storage Tank Actuations

7.3.1.2.9 ~~In-Containment Refueling Water Storage Tank~~ Containment Recirculation

Signals to align the ~~in-containment refueling water storage tank~~ containment recirculation isolation valves are generated from the following conditions:

1. Low-3 in-containment refueling water storage tank water level in coincidence with fourth stage automatic depressurization system actuation (subsection 7.3.1.2.4)

2. Manual initiation

3. Extended loss of ac power sources

There are four parallel containment recirculation paths provided to permit the recirculation of the water provided by the in-containment refueling water storage tank. Two of these paths are provided with two isolation valves in series while the remaining two paths are provided with a single isolation valve in series with a check valve.

Conditions 1 and 2 result in the opening of all isolation valves in all four parallel paths. Condition 3 results in the opening of the two isolation valves that are in series with the check valves.

Condition 1 results from the coincidence of two of the four divisions of in-containment refueling water storage tank water level below the Low-3 setpoint, coincident with an automatic fourth stage automatic depressurization system signal.

Condition 2 consists of two sets of two momentary controls. Manual actuation of both controls of either of the two control sets initiates recirculation in all four parallel paths. A two-control simultaneous actuation prevents inadvertent actuation.

Condition 3 results from the loss of all ac power for a period of time that approaches the 24-hour Class 1E dc battery capability to activate the in-containment refueling water storage tank containment recirculation isolation valves. The timed output holds on restoration of ac power and is manually reset after the batteries are recharged. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four Class 1E battery chargers. Two sensors are connected to each of the four battery charger inputs. The loss of ac power signal is based on the detection of an undervoltage condition by either of the two sensors connected to two of the four battery chargers.

The functional logic relating to activation of the ~~in-containment refueling water storage tank~~ containment recirculation isolation valves is illustrated in Figure 7.2-1, sheets 15 and 16.

Table 7.3-1 (Sheet 8 of 8)

ENGINEERED SAFETY FEATURES ACTUATION SIGNALS

Actuation Signal	No. of Channels/ Switches	Actuation Logic	Permissives and Interlocks
22. Open ~~IRWST~~ Containment Recirculation Valves In Series with Check Valves (Figure 7.2-1, Sheet 15)			
a. Extended undervoltage to Class 1E battery chargers(8)	2/charger	1/2 per charger and 2/4 charger	None
23. Open All ~~IRWST~~ Containment Recirculation Valves (Figure 7.2-1, Sheet 16)			
b. Automatic reactor coolant system depressurization (fourth stage)			(See items 3d through 3f)
Low IRWST level (Low-3 setpoint)	4	2/4 BYP[1]	None
c. Manual initiation	4 switches	2/4 switches	None

Table 7.3-3 (Sheet 1 of 2)

SYSTEM-LEVEL MANUAL INPUT TO THE
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

Manual Control	To Divisions				Figure 7.2-1 Sheet
Manual safeguards actuation #1	A	B	C	D	2 & 11
Manual safeguards actuation #2	A	B	C	D	2 & 11
Manual chemical and volume control system isolation #1	A		C	D	6
Manual chemical and volume control system isolation #2	A		C	D	6
Manual passive residual heat removal heat exchanger alignment ~~actuation~~ #1	A	B		D	8
Manual passive residual heat removal heat exchanger alignment ~~actuation~~ #2	A	B		D	8
Manual steam line isolation #1		B		D	9
Manual steam line isolation #2		B		D	9
Manual steam generator relief isolation #1		B		D	9
Manual steam generator relief isolation #2		B		D	9
Steam/feedwater isolation and safeguards block control #1	A				9
Steam/feedwater isolation and safeguards block control #2		B			9
Steam/feedwater isolation and safeguards block control #3			C		9
Steam/feedwater isolation and safeguards block control #4				D	9
Manual feedwater isolation #1		B		D	10
Manual feedwater isolation #2		B		D	10
Manual steam dump interlock selector #1		B			10
Manual steam dump interlock selector #2				D	10
Pressurizer pressure safeguards block control #1	A				11
Pressurizer pressure safeguards block control #2		B			11
Pressurizer pressure safeguards block control #3			C		11
Pressurizer pressure safeguards block control #4				D	11
Manual core makeup tank injection actuation #1	A	B	C	D	12
Manual core makeup tank injection actuation #2	A	B	C	D	12
Core makeup tank injection actuation block control #1	A				12
Core makeup tank injection actuation block control #2		B			12
Core makeup tank injection actuation block control #3			C		12
Core makeup tank injection actuation block control #4				D	12
Manual passive containment cooling actuation #1	A	B	C		13
Manual passive containment cooling actuation #2	A	B	C		13
Manual passive containment isolation actuation #1	A	B	C	D	13
Manual passive containment isolation actuation #2	A	B	C	D	13
Manual depressurization system stages 1, 2, and 3 actuation #1 & #2	A	B	C	D	15
Manual depressurization system stages 1, 2, and 3 actuation #3 & #4	A	B	C	D	15
Manual depressurization system stage 4 actuation #1 & #2	A	B	C	D	15
Manual depressurization system stage 4 actuation #3 & #4	A	B	C	D	15
Manual IRWST injection actuation #1 & #2	A	B	C	D	16
Manual IRWST injection actuation #3 & #4	A	B	C	D	16
Manual containment recirculation actuation #1 & #2	A	B	C	D	16
Manual containment recirculation actuation #3 & #4	A	B	C	D	16
Manual control room isolation and air supply initiation #1	A	B	C	D	13
Manual control room isolation and air supply initiation #2	A	B	C	D	13

Table 7.3-3 (Sheet 2 of 2)

SYSTEM-LEVEL MANUAL INPUT TO THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

Manual Control	To Divisions				Figure 7.2-1 Sheet
RCS pressure CVS/PRHR block control #1	A				6
RCS pressure CVS/PRHR block control #2		B			6
RCS pressure CVS/PRHR block control #3			C		6
RCS pressure CVS/PRHR block control #4				D	6
~~RNS~~ **Normal residual heat removal system isolation safeguards block control #1**	A				13
~~RNS~~ **Normal residual heat removal system isolation safeguards block control #2**		B			13
Boron dilution block control #1	A				3
Boron dilution block control #2		B			3
Boron dilution block control #3			C		3
Boron dilution block control #4				D	3
Manual RNS isolation #1 & #3	A	B		D	18
Manual RNS isolation #2 & #4	A	B		D	18

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-17 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Table 3.2-1, "Acceptance Criteria," Items 7.iii and 7.iv: These acceptance criteria do not relate to providing a suitable work space environment for MCR operators. There is nothing in Tier 1, Subsection 2.6.3, that evaluates the adequacy/effectiveness/suitability of illumination levels for the facility or the workstations in the facilities. As part of evaluating a suitable work space environment for the MCR and RSR, there should be an assessment of auditory levels (noise) as well. This comment also applies to Table 3.2-1, "Acceptance Criterion," Item 10.ii. This is Open Item 14.3.3-17.

Westinghouse Response:

Illumination level requirements in the MCR and RSR will be added to DCD Tier 1 Subsection 2.6.5 and DCD Tier 2 Subsection 9.5.3.2 as shown below.

DCD Tier 1 Subsection 2.7.1 and Tier 2 Subsection 9.4.1.1.2 will be revised as shown below to address the concern about auditory levels in the MCR and RSR.

NRC Additional Comments:

The wording as proposed in the original response is confusing. The wording should be revised to more clearly state the design requirement.

Westinghouse should also provide a reference for the illumination values proposed.

Westinghouse Additional Response:

The DCD revision shown below includes revised wording to resolve the confusion resulting from the wording proposed in the initial response to this open item.

The values of 50 foot-candles for normal lighting and 10 foot-candles for emergency lighting meet the requirements of the EPRI Utility Requirements Document, Volume III, Chapter 8, Sections 8.3 and 8.5 and MIL-STD-1472F. These values are consistent with recommendations in the *IESNA Lighting Handbook*.

Design Control Document (DCD) Revision:

DCD Tier 1

2.6.5 Lighting System

Design Description

Add the following 2 items.

5. The normal lighting can provide ~~up to~~ 50 foot candles at the safety panel and at the workstations in the MCR and at the RSW.

6. The emergency lighting can provide ~~up to~~ 10 foot candles at the safety panel and at the workstations in the MCR and at the RSW.

Table 2.6.5-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Add the following 2 items.		
5. The normal lighting can provide ~~up to~~ 50 foot candles at the safety panel and at the workstations in the MCR and at the RSW.	i) Testing of the as-built normal lighting in the MCR will be performed.	i) When adjusted for maximum illumination and powered by the main ac power system, the normal lighting in the MCR provides at least 50 foot candles at the safety panel and at the workstations.
	ii) Testing of the as-built normal lighting at the RSW will be performed.	ii) When adjusted for maximum illumination and powered by the main ac power system, the normal lighting provides at least 50 foot candles at the RSW.
6. The emergency lighting can provides ~~up to~~ 10 foot candles at the safety panel and at the workstations in the MCR and at the RSW.	i) Testing of the as-built emergency lighting in the MCR will be performed.	i) When adjusted for maximum illumination and powered by the six Class 1E inverters, the emergency lighting in the MCR provides at least 10 foot candles at the safety panel and at the workstations.
	ii) Testing of the as-built emergency lighting at the RSW will be performed.	ii) When adjusted for maximum illumination and powered by the six Class 1E inverters, the emergency lighting provides at least 10 foot candles at the RSW.

2.7.1 Nuclear Island Nonradioactive Ventilation System

Design Description

The nuclear island nonradioactive ventilation system (VBS) serves the main control room (MCR), technical support center (TSC), Class 1E dc equipment rooms, Class 1E instrumentation and control (I&C) rooms, Class 1E electrical penetration rooms, Class 1E battery rooms, remote shutdown room (RSR), reactor coolant pump trip switchgear rooms, adjacent corridors, and the passive containment cooling system (PCS) valve room during normal plant operation. The VBS consists of the following independent subsystems: the main control room/technical support center HVAC subsystem, the class 1E electrical room HVAC subsystem and the passive containment cooling system valve room heating and ventilation subsystem. The VBS provides heating, ventilation, and cooling to the areas served when ac power is available. The system provides breathable air to the control room and maintains the main control room and technical support center areas at a slightly positive pressure with respect to the adjacent

rooms and outside environment during normal operations. The VBS monitors the main control room supply air for radioactive particulate and iodine concentrations and provides filtration of main control room/technical support center air during conditions of abnormal (high) airborne radioactivity. In addition, the VBS isolates the HVAC penetrations in the main control room boundary on high-high particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system (VES).

Add the following item.

14. The background noise level in the MCR and RSR does not exceed 65 dB(A) when the VBS is operating.

Table 2.7.1-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Add the following item.		
14. The background noise level in the MCR and RSR does not exceed 65 dB(A) when the VBS is operating.	The as-built VBS will be operated and background noise levels in the MCR and RSR will be measured.	The background noise level in the MCR and RSR does not exceed 65 dB(A) when the VBS is operating.

<u>DCD Tier 2</u>

9.4.1.1.2 Power Generation Design Basis

Main Control Room/Technical Support Center Areas

The nuclear island nonradioactive ventilation system provides the following specific functions:

- Controls the main control room and technical support center relative humidity between 25 to 60 percent
- Maintains the main control room and technical support center areas at a slightly positive pressure with respect to the adjacent rooms and outside environment during normal operations to prevent infiltration of unmonitored air into the main control room and technical support center areas

- Isolates the main control room and/or technical support center area from the normal outdoor air intake and provides filtered outdoor air to pressurize the main control room and technical support center areas to a positive pressure of at least 1/8 inch wg when a high gaseous radioactivity concentration is detected in the main control room supply air duct

- Isolates the main control room and/or technical support center area from the normal outdoor air intake and provides 100 percent recirculation air to the main control room and technical support center areas when a high concentration of smoke is detected in the outside air intake

- Provides smoke removal capability for the main control room and technical support center areas

- Maintains the main control room emergency habitability system passive cooling heat sink below its initial design ambient air temperature limit of 75°F

- Maintains the main control room/technical support center carbon dioxide levels below 0.5 percent concentration and the air quality within the guidelines of Table 1 and Appendix C, Table C-1 of Reference 32.

The background noise level in the main control room does not exceed 65 dB(A) when the VBS is operating.

The system maintains the following room temperatures based on the maximum and minimum outside air safety temperature conditions shown in Chapter 2, Table 2-1:

Area	**Temperature (°F)**
Main control room	67 - 75
Technical support center	67 - 78

Class 1E Electrical Rooms/Remote Shutdown Room

The nuclear island nonradioactive ventilation system provides the following specific functions:

- Exhausts air from the Class 1E battery rooms to limit the concentration of hydrogen gas to less than 2 percent by volume in accordance with Regulatory Guide 1.128 (Reference 31).

- Maintains the Class 1E electrical room emergency passive cooling heat sink below its initial design ambient air temperature limit of 75°F

- Provides smoke removal capability for the Class 1E electrical equipment rooms and battery rooms

The background noise level in the remote shutdown room does not exceed 65 dB(A) when the VBS is operating.

The system maintains the following room temperatures based on the maximum and minimum outside air safety temperature conditions shown in Chapter 2, Table 2-1:

Area	Temperature (°F)
Class 1E battery rooms	67 - 73
Class 1E dc equipment rooms	67 - 73
Class 1E electrical penetration rooms	67 - 73
Class 1E instrumentation and control rooms	67 - 73
Corridors	67 - 73
Remote shutdown room	67 - 73
Reactor coolant pump trip switchgear rooms	67 - 73
HVAC equipment rooms	50 - 85

9.5.3.2.1 Normal Lighting

Power to the normal lighting system is supplied from the non-Class 1E ac power distribution system at the following voltage levels:

- 480/277 V, three-phase, four-wire, grounded neutral system lighting panels are fed from the 480 V motor control centers; this source is for the lighting fixtures rated at 480/277 V and for the welding receptacles.

- 208/120 V, three-phase, four-wire, grounded neutral system distribution panels are fed from the 480 V motor control centers through dry-type 480-208/120 V transformers; this source is for lighting and utility receptacles.

- 208/120 V, three-phase, four-wire, grounded neutral regulated power fed from the 480 V motor control centers through the Class 1E 480 - 208/120 V voltage regulating transformers (divisions B and C); this source is for the normal and emergency lighting in the main control room and remote shutdown room and is isolated through two series fuses for isolation. The normal lighting in these plant areas is non-Class 1E.

The normal lighting system has the following features:

- The normal lighting system is powered from the diesel-backed buses and the lighting load is distributed between the two onsite standby diesel generator buses.

- The motor control centers powering the normal lighting system are energized from the 480 V load centers connected in a tie-breaker configuration.

- Lighting distribution panel branch circuit breakers are controlled by a lighting control system. Approximately 75 percent of the normal lighting is tripped off automatically upon loss of normal ac power (except in the main control room and in the remote shutdown room) to limit the load on the onsite standby diesel generators. The lighting control system allows the operator to energize or de-energize lighting in selected areas based on the actual need and available power from the onsite standby diesel generators.

- The lighting circuits are staggered as much as practical. The staggered circuits receive power from separate buses to prevent complete loss of light in the event of a bus or a circuit failure.

- The lighting fixtures located in the vicinity of safety-related equipment are supported so that they do not adversely impact this equipment when subjected to the seismic loading of a safe shutdown earthquake.

- **The control room and remote shutdown room lighting utilizes semi-indirect, low-glare lighting fixtures and programmable dimming features. The normal control room lighting provides at least 50 foot candles of illumination at the safety panel and at the workstations when the dimming features are adjusted for maximum illumination. The normal remote shutdown room lighting provides at least 50 foot candles of illumination at the remote shutdown workstation when the dimming features are adjusted for maximum illumination.**

9.5.3.2.2 Emergency Lighting

Emergency lighting is designed to provide the required illumination levels in the areas as described below:

- The main control room and remote shutdown room each has emergency lighting consisting of 120 V ac fluorescent lighting fixtures which are continuously energized. The fixtures are powered from the Class 1E 125 V dc switchboards through the Class 1E 208Y/120 V ac inverters and are isolated through two series fuses. Three hour fire barrier separation is provided between redundant emergency lightning power supplies and cables outside the main control room and the remote shutdown area. The control room lighting complies with the human factor requirements by utilizing semi-indirect, low-glare lighting fixtures and programmable dimming features. The control room emergency lighting is integrated with normal lighting that consists of identical lighting fixtures and dimming features. The emergency lighting system is designed so that, to the extent practical, alternate emergency lighting fixtures are fed from separate divisions of the Class 1E dc and uninterruptible power supply system. Both normal and emergency lighting fixtures, controllers, dimmers, and associated cables used in the main control room and remote shutdown room are non-Class 1E. The ceiling grid network, raceways and fixtures utilize seismic supports. A single fault cannot interrupt all of the lighting in the main control room and at the remote shutdown workstation simultaneously. **The emergency lighting provides at least 10 foot candles of illumination at the safety**

panel, at the workstations in the control room, and at the remote shutdown workstation when the dimming features are adjusted for maximum illumination.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **17.3.2-2**

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

Summary of Issue:

Implementation of QA Program for AP1000 Design

Westinghouse stated that a project-specific quality control plan was used to implement the requirements of the Westinghouse QMS program. The staff plans to conduct an inspection of the implementation of the project-specific quality plan to verify that design activities conducted for the AP1000 project complied with the Westinghouse QMS and the requirements of 10 CFR Part 50, Appendix B. As discussed in this report Chapter 20, "Generic Issues," the NRC staff will also address the implementation of QA requirements 10 CFR 50.34(f)(3) and NUREG-0933, Item I.F.2, during this inspection. This is DSER Open Item 17.3.2-2.

Westinghouse Response:

The AP1000 DCD Chapter 17 addresses the Quality Assurance program for the AP1000. Westinghouse is currently interfacing with the NRC to schedule the inspection.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

DSER Open Item Number: 17.3.2-3 (Response Revision 1)

Original RAI Number(s): 260.007

Summary of Issue:

Exception to RG 1.28: As noted previously in this chapter, in DCD Tier 2 Section 1, Appendix 1A, the applicant took exception to record retention recommendations in RG 1.28. Specifically, RG 1.28, Regulatory Position C.2, Quality Assurance Records, states, in part, that programmatic nonpermanent records should be retained for at least 3 years. For programmatic nonpermanent records, the retention period should be considered to begin upon completion of the activity. In addition, RG 1.28 states that product and programmatic nonpermanent records should be retained at least until the date of issuance of the full power operating license of the unit. Under 10 CFR Part 52, issuance of a COL is comparable to issuance of a full power operating license under 10 CFR Part 50. The applicant stated that because a definitive schedule for obtaining a full power operating license does not exist, the records retention plan is keyed to the final design approval. The applicant stated that a 3 year programmatic records retention period will be initiated starting on the date that NRC issues an AP1000 final design approval. The NRC staff determined that this exception to RG 1.28 may not be acceptable since programmatic nonpermanent records could be discarded 3 years after issuance of a final design approval; therefore, these records may not be available to a future COL applicant. The NRC staff requested additional information to assess the basis for not retaining nonpermanent records until a COL is issued. The applicant should provide a list of the specific records types that they are proposing to discard after 3 years. The applicant should also provide additional justification for discarding each of these record types after final design approval. This information was requested from the applicant through RAI 260.007. This is DSER Open Item 17.3.2-3.

Westinghouse Response:

This question was originally identified as RAI OI-260.007 Rev. 0. Westinghouse provided a response to RAI OI-260.007 Rev. 0, which was originally transmitted to the NRC via DCP/NRC1588 dated 05/13/03.

NRC Additional Comments:

Westinghouse is requested to look at their response as compared to the guidance of Table 1 in RG 1.28.

Westinghouse Revised Response:

Westinghouse has re-reviewed the guidance of Table 1 in RG 1.28 as compared to our procedures, and has determined that we conform to the guidance. The DCD will be revised as stated below in the "Design Control Document (DCD) Revision" portion of this response.

Design Control Document (DCD) Revision:

Revise DCD Section 1, Appendix 1A as follows:

Reg. Guide 1.28, Rev. 3, 8/85 - Quality Assurance Program Requirements (Design and Construction)

2.	Criteria 17 10 CFR 50 Appendix B	~~Exception~~**Conforms**	~~Section 2, Quality Records requires that programmatic nonpermanent records be retained for 3 years. An additional requirement states that programmatic nonpermanent records shall be retained "at least until the date of issuance of a full-power operating license of the unit." As a definitive schedule for obtaining a full-power operating license does not exist, Westinghouse will follow a records retention plan that is keyed to the Final Design Approval. Compliance will be accomplished by initiating a retention period of 3 years for programmatic nonpermanent records starting on the date that NRC issues AP1000 a FDA.~~

PRA Revision:

None

DSER Open Item Number: 17.5-1

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.

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.**

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: 18.11.3.5-1 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Criterion 3, "Plant Personnel", states that participants in validation tests should represent an unbiased sample; be representative of actual plant personnel; reflect characteristics of the population of plant personnel; include shift supervisors, reactor operators, shift technical advisors, etc., and minimum and normal crew configurations. Westinghouse submitted WCAP-14396 (Revision 3), "Man-In-The-Loop Test Plan Description," Section 2.4.3, "Subjects," address the composition of the "target user population," the test subject population. While this WCAP addresses preliminary or "engineering" tests, rather than final or "validation" tests, with validation tests addressed by WCAP-15860, the test subject selection criteria should be applicable for both test types. The applicant should amplify/clarify or explain how validation tests address this NUREG-0711 item. Therefore, this is Open Item 18.11.3.5-1.

NRC Additional Comments:

The original response to this open item contains a confusing reference. Westinghouse should revise the response to address this issue.

Westinghouse Response:

The following response is a complete response to this open item. The reference problem has been corrected.

Section 2.4.3, "Subjects," in WCAP-14396 (Revision 3), "Man-in-the-loop Test Plan Description" applies only to preliminary or "engineering" tests, and is not applicable to final validation tests. To address the topic for validation tests, the following section will be added to WCAP-15860 (Rev. 1), "Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan":

4.9 SUBJECTS

In actual operation, the AP1000 main control room and associated HSID features will be used only by highly trained, qualified commercial nuclear power plant (NPP) operating crews. The hypothetical group of all such qualified crew members is referred to here as the "target user population". To assure that test subjects will represent this population, validation crews will be comprised of currently qualified operating crews, as adjusted in number to man the AP1000 control room for conditions of minimum and maximum staffing. This excludes, by definition, members of the design organization.

The target users can be subdivided on the basis of qualification and experience. For AP1000, two subgroups of interest are referred to here as "operators" and "supervisors". Supervisors by definition have longer experience and higher qualifications. To ensure a conservative test, steps will be taken to identify and select test subjects from crews with less experience or unexceptional performance. This may be difficult to achieve for several reasons, include the sensitivity of being identified as average or below. However, test subjects will in no case be selected for their superior skills or experience.

A key question is the number of subjects to be used in each test (i.e., sample size = n). Several authors have examined the mathematical models that underlie descriptive usability evaluations. Plotting the proportion of usability problems detected as a function of number of test participants, the relation can be modeled as a simple Poisson process. In essence, each successive test subject tends to reveal fewer findings. Reference [13] continues in this vein to suggest that five test subjects are typically enough to detect 70% to 90% of major usability problems in a prototype. Thus, a minimum of n = 6 subjects (3 crews) is proposed as sufficient for validation tests.

Prior to testing, subjects will receive a week of training on the testbed. Training content will exclude actual scenarios used for validation testing. Training should be sufficient to prepare subjects for the demands of the planned tests. However, since the training is relatively brief, it is not expected to produce test subjects for the new design quite equal to the fully qualified and experienced crew of an existing plant. As a result, inexperience still tends to weigh against favorable validation results, but as a conservative error, this is acceptable.

The following changes will be made to section 7, References, of WCAP-15860, "Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan":

7 REFERENCES

1. ANSI HFS-100-1988, "American Standard for Human Factors Engineering of Visual Display Terminal Workstations." American National Standards Institute, Santa Monica, California, 1988.

2. CEI/IEC 964 "Design for Control Rooms of Nuclear Power Plants." International Electrotechnical Commission, Geneva, Switzerland, 1989.

3. DOD-HDBK-76 I A "Human Engineering Guidelines for Management Information Systems." US Department of Defense, Office of Management and Budget, Washington, D.C., 1990.

4. IEEE Std. 845-1999 "IEEE Guide for the Evaluation of Human-System Performance in Nuclear Power Generating Stations." Institute of Electrical and Electronics Engineers, 1999.

5. NUREG-0899 "Guidelines for the Preparation of Emergency Operating Procedures." US Nuclear Regulatory Commission, Washington, DC., August 1982.

6. NUREG-1358 "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures." US Nuclear Regulatory Commission, Washington, DC., April 1989.

7. NUREG-0711 "Human Factors Engineering Program Review Model." US Nuclear Regulatory Commission, Washington, D.C., July 1994.

8. NUREG-0700 "Human-System Interface Design Review Guideline," Rev. 1. US Nuclear Regulatory Commission, Washington, DC., June 1996.

9. NUREG/CR-5908 "Advanced Human-System Interface Design Guidelines." US Nuclear Regulatory Commission, Washington, DC., July 1994.

10. NUREG/CR-6501 "Human Factors Engineering Guidelines for the Review of Advanced Alarm Systems." US Nuclear Regulatory Commission, Washington, DC., September 1994.

11. Regulatory Guide I .33, "Quality Assurance Program Requirements" Revision 2, US Nuclear Regulatory Commission Washington, D.C.

12. ANSI/ANS-3.5-1993, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," approved March 29, 1993.

13. **Virzi, R. A. "Refining the test phase of usability evaluation: How many subjects is enough?," *Human Factors*, 34(4), 1992, 457-468.**

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **19.2.6-1 (Revision 1)**

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

Summary of Issue:

Deterministic Containment Capacity

The evaluation of ultimate capacity of the AP1000 containment is presented in DCD Tier 2 Section 3.8.2.4.2. In this section, the applicant evaluates the containment capacity at Service Level C limit by examining various parts of the containment structure, cylindrical shell, top and bottom heads, equipment hatches and covers, personnel airlocks, and mechanical and electrical penetrations. At Service Level C, the applicant determined that the capacity of the ellipsoidal head is 627 KPa (91 psig) at 149° C (300 °F) and the capacity of the equipment hatch covers is 558 KPa (81 psig) at 149 ° C (300 °F) using NE 3222. Using Code Case N284, the capacity of the equipment hatch covers was determined to be 834 KPa (121 psig) at 149 ° C (300 ° F). The staff has always maintained that the provisions of Code Case N284 apply to local buckling cases only. The equipment hatch cover buckling is a global buckling phenomenon and therefore, the use of Code Case N284 is not appropriate. The Service Level C capacity of the AP1000 containment structure should be the lowest value, 558 KPa (81 psig) at 149_C (300 ° F). In Section 42.3.1 of the PRA, the applicant states, "The 90 psig [620 KPa] is the Service Level C containment failure pressure at 300_F." The staff does not agree with this assessment. The applicant should address why 558 KPa (81 psig) at 149 ° C (300 ° F) is not the limiting severe-accident pressure for the AP1000 containment. This is Open Item 19.2.6-1

Westinghouse Response (Revision 1):

Westinghouse has evaluated the equipment hatch cover meeting the requirements for Service Level C. Westinghouse satisfies the 10CFR50.34 that requires "meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE - 3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone." Westinghouse fulfills the commission policy related to Service Level C requirements in SECY-93-087 that requires meeting Level C limits "under the more likely severe accident challenges". This is discussed in more detail below.

<u>10CFR50.34 Requirements</u>

Westinghouse has performed a stress analysis of the AP1000 containment structure following the requirements set forth in 10CFR50.34 noting that evaluation of instability is not required:

> (A)(1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE - 3220, Service Level C Limits, except that evaluation of instability is not

> required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subsubarticle CC - 3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

The maximum internal pressure for the containment vessel satisfying the ASME Service Level C stress intensity limits (without consideration of buckling instability) is 117.2 psig at the 300°F design temperature. The results of the stress analysis show that the maximum pressure is governed by the circumferential membrane stress in the cylindrical shell. This pressure is above the 90 psig pressure that is calculated in accordance with 10CFR 50.34 for Service Level C severe accident phenomena evaluation that includes hydrogen burn. **Therefore,**

> **Capacity of hatch = 117.2 psig > 90 psig with hydrogen burn OKAY**

SECY-93-087 Requirements

Westinghouse meets the requirements set forth in SECY-93-087 that requires that buckling be considered. The maximum containment pressure is based on the more likely severe accident challenges as stated in the commission policy statement given in SECY-93-087 under Containment Performance:

> **Therefore, the Commission approves the staff's position to use the following deterministic containment performance goal in the evaluation of the passive ALWRs as a complement to the CCFP approach approved by the Commission in its SRM of June 26, 1990:**
>
> > **"The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containments stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) for approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products."**

The capacity of the equipment hatch covers were calculated to be 81 psig at 300°F considering buckling following ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE – 3220 requirements.

Most of the likely severe accident scenarios result in containment pressures less than the design pressure. Severe accident challenges are addressed with engineered features to mitigate them in the AP1000. Passive containment cooling water is highly reliable, and failure of water cooling only occurs in a small fraction of the core damage frequency. Combustible gases are controlled with glow plug igniters. External reactor vessel cooling is expected to maintain molten core debris inside the reactor vessel.

A bounding severe accident scenario is described in PRA report Chapter 40 subsection 40.4.2. The nominal case described results in a containment pressure of 81 psig at 24 hours after the initiation of core damage. This equals the Service Level C capacity of the hatch cover under ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE – 3220 requirements. After 24 hours, the pressurization in this bounding analysis produces a peak pressure of 105 psig, which has a small probability, 0.022, of failure. Given that the frequency of this severe accident sequence, including containment failure, is on the order of 10^{-15} per reactor-year, and that this analysis bounds the more likely severe accident challenges, the SECY-93-087 requirement is met.

A change is made in Section 42.3.1 of the PRA for clarification related to maximum pressure.

Design Control Document (DCD) Revision:

None

PRA Revision:

Revise Section 42.3.1

It is noted that a containment conditional failure probability distribution for a containment temperature at 331°F which corresponds to saturation at 90 psig is also developed. This distribution is referenced in the discussion on passive containment cooling system (PCS) failure and fission-product release category CFL (see Chapters 34 and 45). ~~The 90 psig [620 KPa] is the Service Level C containment failure pressure at 300_F.~~ **The 90 psig [620 Kpa] is the maximum pressure that is calculated in accordance with 10CFR 50.34 for the severe accident phenomena for Service Level C evaluation that includes hydrogen burn.**

A maximum containment pressure of 81 psig [559 Kpa] is calculated at 24 hours following the onset of core damage for the bounding severe accident phenomena as described in subsection 40.4.2. This bounds the more likely severe accident challenges that are required to be evaluated against Service Level C in accordance with SECY-93-087 Requirements.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-1 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

<u>Deterministic Strength Factor</u>

The deterministic design process involves the use of: (1) actual stress that is less than the allowable value specified in the design code, and (2) the margin used in the code allowable values by the code or standard developing body. The applicant has not explained how this factor was used in its probabilistic fragility analysis. This is Open Item 19A.2-1.

Original Westinghouse Response:

Deterministic strength factors are defined by a margin factor based on the failure that is being evaluated for the structural element being analyzed. These margin factors are established based on the controlling load combination, associated allowables, margin of actual stress to code allowable, and the factor of safety related with the code allowable. The same margin factors are used for the AP1000 plant as employed for the AP600 plant. The NRC, as part of the AP600 licensing process, reviewed these margin factors. The deterministic margin factors (X_i) are used in the formula given in Probabilistic Fragility Analysis item of AP1000 PRA section 55.2.2.3, Analysis of Structure Response that defines the mean peak seismic ground capacity. This is also discussed in AP600 FSER 19A.2-1

The mean peak seismic ground capacity, A_m, is related to the stress and strength design margin factors by the following expression:

$$A_m = (\Pi_i [X_i])A_o$$

where,

A_m = Mean peak seismic ground capacity
X_i = ith design mean margin factor
Π_i = Product notation
A_o = Nominal seismic peak ground capacity

Additional NRC Comment:

Westinghouse is to provide one specific example that demonstrates the use of deterministic strength factors.

Westinghouse Response to Additional NRC Comment:

In a meeting with the NRC staff on July 10, 2003, Westinghouse agreed to provide an example that explains how deterministic strength factors, variable strength factors, and material factors

are used. This response addresses the deterministic strength factor. See Revision 1 responses to DSER OI 19A.2-2 and 19A.2-3 for variable strength and material strength examples. The three factors of deterministic strength, variable strength, and material define the strength seismic margin fragility data. The other factors in the HCLPF calculation are related to demand. The strength seismic margin factors are combined with the demand seismic margin factors to define the HCLPF values.

The example that is shown is the steam generator column buckling HCLPF calculation.

The deterministic strength margin factors have no variability and are defined based on the code allowable and the factor of safety applied to the code allowable. In the example being presented, steam generator column buckling, the code allowable buckling load is defined as two-thirds of the critical buckling load. The factor of safety is 1.5. Therefore, the deterministic strength factor can be defined:

Factor = [1.5 x Allowable Load – Dead Weight] / SSE Column Load

Allowable Load = 5084 kips
Dead Weight = 2086 kips
SSE Column Load = 1841 kips

Factor = [1.5 x 5084 – 2086] / 1841 = 3.0

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-2 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Variability exists between the design capacity and the test capacity. This phenomenon is inherent in the manner in which an actual structure redistributes loads based on redundancy, excess capacity provided by design, end constraints and other factors. The applicant has not explained how this factor was used in its probabilistic fragility analysis. This is Open Item 19A.2-2.

Original Westinghouse Response:

Variable strength factors are defined by a margin factor and lognormal standard deviations (LSDs). They are used in the formulas given in Probabilistic Fragility Analysis item of AP1000 PRA section 55.2.2.3, Analysis of Structure Response. The NRC, as part of the AP600 licensing process, reviewed these strength factors and LSDs. The same strength margin factors and LSDs are used for the AP1000 plant as employed for the AP600 plant. NRC AP600 Open Item 720.447F and AP600 FSER 19A.2.1.2.1 addressed these factors with LSD recommendations. The NRC recommends that variable strength composite LSDs used for certain primary component supports be increased to be consistent with values given in EPRI TR-103959. The components addressed in these open items are shown below as examples of the variable strength factors used for the AP1000 plant. Westinghouse used the standard deviations (LSD) with an associated margin factors based on the failure that is being evaluated for the structural element being analyzed.

RPV High-Strength Bolts

Failure is based on shear with the margin factor equal to the ratio of the bolt shear strength to the allowable shear strength that is equal to 0.42 of the ultimate stress. A LSD value of 0.10 is used with this margin factor consistent with AP600 FSER 19A.2.1.2.1.

Pressurizer Upper Support Strut Weld

Failure is based on shear with the margin factor equal to the ratio of the weld shear strength to the allowable shear strength that is equal to 0.42 of the ultimate stress. A LSD value of 0.19 is used with this margin factor consistent with AP600 FSER 19A.2.1.2.1.

Steam Generator Upper Support Ring Girder Flange Bolt

The AP1000 plant does not have an upper support ring girder, and therefore this structural support component does not apply to the AP1000 plant.

Additional NRC Comment:

Westinghouse is to provide one specific example that demonstrates the use of variable strength factors.

Westinghouse Response to Additional NRC Comment:

In a meeting with the NRC staff on July 10, 2003, Westinghouse agreed to provide an example that explains how deterministic strength factors, variable strength factors, and material factors are used. This response addresses the variable strength factor. See Revision 1 responses to DSER OI 19A.2-1 and 19A.2-3 for deterministic strength and material strength examples. The three factors of deterministic strength, variable strength, and material define the strength seismic margin fragility data. The other factors in the HCLPF calculation are related to demand. The strength seismic margin factors are combined with the demand seismic margin factors to define the HCLPF values.

The example that is shown is the steam generator column buckling HCLPF calculation.

Variability exists in the strength of the structural component. This is a function of the failure mode as well as the actual strength versus the theoretical or code strength. For this example, variable strength factors consider the actual strength (measured by tests) versus calculated theoretical strength on which the code allowable is based, the variability in the theoretical calculated strength due to assumed design conditions (e.g., boundary conditions), and variability due to fabrication. Reference 19A.2-2-1 is used to define the seismic margin parameters (strength factors and standard deviation).

Actual versus theoretical strength:	Strength Factor = 1.005 and standard deviation = 0.093
Design Conditions:	Strength Factor = 1.01 and standard deviation = 0.04
Fabrication:	Strength Factor = 1.0 and standard deviation = 0.05

The above factors are combined to define the total variable seismic margin factors.

Total Variable Strength Factor = 1.005 x 1.01 = 1.015

$$\text{Standard deviation} = [0.093^2 + 0.04^2 + 0.052]^{1/2} = 0.11$$

Reference:

19A.2-2-1 Bjorhovde, Reidar, T. V. Galambos, and M. K. Ravindra, "LRFD Criteria for Steel Beam-Columns," Journal of the Structural Division, ASCE, No. ST9, Paper 14016, Sept 1978, pp. 1371-1387.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

DSER Open Item Number: 19A.2-3 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Material

The allowable stress values provided in codes and standards are based on minimum specified yield strength in tension or compressive strength in crushing. Consequently, actual material properties that are derived from the yield strength or crushing strength have variability. The applicant has not explained how this factor was used in its probabilistic fragility analysis. This is Open Item 19A.2-3.

Original Westinghouse Response:

Material strength factors are defined by margin factors and lognormal standard deviations (LSDs). They are used in the formulas given in Probabilistic Fragility Analysis item of AP1000 PRA section 55.2.2.3, Analysis of Structure Response. The same material margin factors and LSDs are used for the AP1000 plant as employed for the AP600 plant. The NRC, as part of the AP600 licensing process, reviewed these material factors and LSDs and found them acceptable (FSER 19A.2.1.2.1).

Additional NRC Comment:

Westinghouse is to provide one specific example that demonstrates the use of material strength factors.

Westinghouse Response to Additional NRC Comment:

In a meeting with the NRC staff on July 10, 2003, Westinghouse agreed to provide an example that explains how deterministic strength factors, variable strength factors, and material factors are used. This response addresses the material strength factor. See Revision 1 responses to DSER OI 19A.2-1 and 19A.2-2 for deterministic strength and variable strength examples. The three factors of deterministic strength, variable strength, and material define the strength seismic margin fragility data. The other factors in the HCLPF calculation are related to demand. The strength seismic margin factors are combined with the demand seismic margin factors to define the HCLPF values.

The example that is shown is the steam generator column buckling HCLPF calculation.

The expected minimum material properties are used to calculate the code allowable stress or load. Like the variable strength factors, these factors are dependent on the failure mode being

investigated. For column buckling, the material property of significance is modulus of elasticity. Using Reference 19A.2-3-1, the associated seismic margin factors are:

Material Strength Factor = 1.0

Standard deviation = 0.06

Reference:

19A.2-3-1 Galambos, T.V., "Load and Resistance Factor Design," Engineering Journal, American Institute of Steel Construction, Third Quarter, 1981.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-4 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Analysis of Modeling Error

Modeling error stems from a number of sources that include stiffness parameters, modeling of masses due to live load, connectivity between structural members, support conditions, and others. The applicant did not explain how this factor was used in its probabilistic fragility analysis of various structures and equipment. For the modal frequency variation, the applicant used a composite logarithmic standard deviation, βc, of 0.3. The use of a βc value of 0.3 means that modal frequency values can vary by a factor of 1.8. The applicant needs to justify the use of such a high variability factor for the natural frequency calculations when using detailed finite element models. This is Open Item 19A.2-4.

NRC Follow-On Comments:

Allowing the modal frequency values to vary by a factor of 1.8 is too conservative. The composite standard deviation of 0.3 as provided in ASCE document for uncertainty in seismic analysis and design of nuclear facilities should not apply to nuclear facilities like the AP1000 because of the extensive analysis and construction QA applied. Westinghouse should show how conservative this factor is.

Westinghouse Response (Revision 1):

~~As stated in AP1000 PRA Chapter 55, modeling errors are related to: Mode Shapes; Modal Frequency Variability; and Imperfections. The fragility data for the modeling errors are defined by margin factors and lognormal standard deviations (LSDs). They are used in the formulas given in Probabilistic Fragility Analysis item of AP1000 PRA section 55.2.2.3, Analysis of Structure Response. This is the same procedure and methodology as used for the AP600 plant. Further, the same modeling error margin factors and LSDs are used for the AP1000 plant as utilized for the AP600 plant. The NRC, as part of the AP600 licensing process, reviewed the methodology and values used for the modeling fragility data and found them acceptable (documented in AP600 FSER 19A.2.1.1.3).~~

Westinghouse agrees that the modal frequency can vary by a factor of 1.8 **when using a composite standard deviation of 0.3. In the AP600 and AP1000 seismic margin analyses** t~~T~~he modal frequency variation is based on the recommendations of Reference 19A.3-1-1. It is stated in this reference (Chapter 5, page 144 and 145): "The modal frequency variability shifts the frequency which spectral accelerations are to be determined, ... Based on experience and Hadjian et al., 1977, the coefficient of variation (approximate logarithmic standard deviation) of frequency is estimated to be approximately 0.3." ~~This variability factor was reviewed by the~~

~~NRC and documented as acceptable in FSER 19A.2.1.1.3 Modeling, Modal Frequency Variability.~~ This variation is too conservative for the AP1000.

A recognized factor is 1.3 (i.e., ±0.15 variation of the frequency) based on the recommended maximum broadening of the floor response spectra peaks associated with the structural frequencies as given in USNRC Regulatory Guide 1.122. This variation is used in broadening the AP1000 floor response spectra as described in DCD subsection 3.7.2-5. This factor was defined accounting for the various uncertainties in structural frequencies due to such items as: material properties of the structure and soil, damping values, soil-structure interaction (for the AP1000 rock structure-interaction), and approximations in modeling techniques used in the seismic analyses. This is consistent with a standard deviation of 0.14. For seismic margin where the review level earthquake is 0.5g, this factor is also valid for steel structures such as the steel containment vessel.

For concrete structures the 1.3 factor is low since the higher seismic levels will cause more cracking and will reduce the structural frequencies. Tests have been performed on shear walls that have demonstrated lower frequency response due to concrete cracking. Tests on concrete filled steel plate modules (Akiyama et al, SMIRT 1989) also show a reduction in frequency as well as an increase in damping as the magnitude of response increases. Criteria have been presented in FEMA documents (FEMA 356, Table 6-5, and 274, Section C6.4.1.2). Effective stiffnesses for concrete walls and diaphragms are given in the table below (Table 19A.2-4-1) that is derived from FEMA 356, Table 6-5. Using these criteria, a reduction in structural frequency can be as much as 20% from that predicted in the AP1000 SSE analyses using 0.80 times the gross (uncracked) section properties;

$$F_{cr} = F_{gr} \times [0.5E_cI_g / 0.8E_cI_g]^{½} = 0.8\ F_{gr}$$

F_{gr} = structural peak frequency based on gross section properties
F_{cr} = structural peak frequency based on cracked section properties

Table 19A.2-4-1 – Effective Stiffness of Concrete Walls and Diaphragms

Member	Flexural Rigidity	Shear Rigidity
Walls and Diaphrams – uncracked , $f_b<f_{cr}$, $V<V_c$	$0.8E_cI_g$	$0.8G_cA_w$
Walls and Diaphrams – cracked, $f_b>f_{cr}$, $V>V_c$	$0.5E_cI_g$	$0.5G_cA_w$

E_c = concrete compressive modulus G_c = concrete shear modulus = $0.4E_c$ E_c = concrete modulus I_g = gross moment of inertia	A_w = web area A_s = gross area of the reinforcing steel f_b = bending stress f_{cr} = cracking stress	V = wall shear V_c = nominal concrete shear capacity

Therefore, recognizing that Westinghouse is using cracked concrete section properties (0.80 times gross section properties) in the models used to generate floor response spectra for the SSE of 0.3g, and that the review level earthquake is 0.5g causing more cracking, the margin factor on concrete structural frequencies is taken as 1.35 (-0.20 to +0.15 on structural peak frequencies).

Chapter 55 of the AP1000 PRA will be modified to reflect the removal of the conservatism.

References

19A.3-1-1 Uncertainty and Conservatism in the Seismic Analysis and Design of Nuclear Facilities, Working Group on Quantification of Uncertainties, American Society of Civil Engineers, ISBN 0-87262-547-8, 1986.

Design Control Document (DCD) Revision:

None

PRA Revision:

~~None~~**From AP1000 PRA Revision 3, Chapter 55, Section 55.2.2.3, page 55-7:**

Modal Frequency Variability

Shifts in the frequency affect spectral acceleration levels and introduce error. **For steel structures** ~~T~~this is reflected in the seismic margin analysis by using a log-normal standard deviation calculated as the ratio of the spectral acceleration value associated with a one-sigma variation in frequency, and the spectral acceleration value at the median centered frequency.

$$(\beta_c)_f = \ln\{S_\beta/S_f\}$$

where,

S_β = Spectral acceleration value at the 84% exceedance probability frequency estimate, f_β
S_f = Spectral acceleration value at median centered frequency
f = median centered frequency
f_β = 84% exceedance probability frequency estimate = $f \times e^{[(+/-)0.314]}$

This is equivalent to a variation of ±15% on the peak frequency of the steel structures. For concrete structures a variation on the structural frequency is –20% to + 15% on the peak frequency of the concrete structures.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-5 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Soil-Structure Interaction

How the structure behaves with the foundation material in which the structure is embedded when subjected to seismic excitation, is analytically determined by the soil-structure interaction (SSI) analysis. For design purposes, the soil parameters are varied by a factor of 2 higher and lower, then the results are enveloped. Consequently, the SSI effect can introduce a considerable variation in the calculated margin. However, the API000 design is to be located on hard rock sites and no SSI analysis is involved in its design. Therefore, the discussion about the SSI related variability in Chapter 55 of the PRA report for AP1000 is inappropriate, since the use of the variability factor $(\beta_c)_{SSI}$, is not justified. This issue is Open Item 19A.2-5.

NRC Follow-On Comment:

Westinghouse should remove the SSI related variability from Chapter 55 or do more to demonstrate that the SSI related variability results in smaller HCLPF values.

Westinghouse Response (Revision 1):

Westinghouse agrees that soil structure interaction is not applicable for the AP1000 plant design that is being licensed for hard rock sites. ~~The SSI related variability in Chapter 55 of the AP1000 PRA report would be applicable for an AP1000 on soil sites.~~ **AP1000 PRA Chapter 55 will be revised to remove soil structure interaction.**

Design Control Document (DCD) Revision:

None

PRA Revision:

~~None~~From AP1000 PRA Revision 3 Chapter 55, Section 55.2.2.3 , page 55-4:

The conservatisms and variability identified and considered in this assessment are associated with stress and strength margin factors. The basic grouping of margin factors are: deterministic strength factor; variable strength factors; material; damping; inelastic energy absorption, ductility; and analysis or modeling error~~; and soil structure interaction~~. These margin factors are discussed below.

From AP1000 PRA Chapter 55, Section 55.2.2.3, page 55-7:

~~**Soil Structure Interaction**~~

~~In the design of the AP1000 plant, envelope spectra will be used to reflect the different soil conditions. No credit is taken in the development of the HCLPF values recognizing that the specific site seismic requirements can be much smaller. Variability, $(\beta_e)_{SSI}$, is estimated to be 0.1.~~

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-6 (Response Revision 1)

Original RAI Number(s): n/a

Summary of Issue:

The applicant used the CDFM method to calculate the HCLPF value of the shield building using strength, inelastic energy absorption and damping as areas where the shield building capacity is increased over the design capacity to determine the cumulative effect of those factors. The applicant has increased the shear capacity of a concrete section by increasing the shear modulus to account for the shear strength of reinforcement bars where the shear load exceeds the shear strength of concrete alone. The ACI 349 Code, the applicable concrete design code, allows the addition of reinforcement strength, but not by increasing the shear modulus of the concrete section. The shield building tension ring has a HCLPF value of 0.51g (Table 55-1, Sheet 1 of 4). Therefore, a validation of the capacity of the shield building shear walls is important. With respect to inelastic energy absorption and damping factors, it is not clear as to whether or not the applicant has double counted damping values through the use of hysteretic damping for inelastic energy absorption and a damping value of 10 percent. The applicant needs to justify the details of the CDFM approach for calculating HCLPF values for important structures and equipment. It should be noted that the containment internal structure and the nuclear island basemat are predicted to lift up under the SSE loading. As noted in Section 3.7 of this report, the effect of uplift due to design basis seismic excitation is an open area. Consequently, at 0.5g review level earthquake, the capacity of the tension ring could potentially be lower. Therefore, the validation of HCLPF values calculated by the CDFM approach is Open Item 19A.2-6.

Westinghouse Response (Revision 1):

Shear Modulus

The shear modulus was not modified to increase shear capacity, but to reflect the section properties with cracking. The analysis of the Shield Building Roof is performed in steps starting with a monolithic stiffness associated with an uncracked section. Based on the results from analysis using this structural configuration (no cracking), section properties were modified to reflect cracking. Using the results obtained that account for concrete cracking and redistribution of the load within the Shield Building Roof, structural margin for the critical sections (columns, tension ring, at PCS Tank Location, PCCS tank walls) were calculated based on the maximum load that these critical elements can transfer without failure.

The total section shear strength is calculated with contributions of concrete and reinforcement together. The total in-plane shear strength including concrete and rebars contributions uses Reference 19A.2-6-1, Appendix L, "Shear Strength of Concrete Walls". Taking into account the axial compression, the total out-of-plane shear concrete strength is based on beam action behavior.

The PRA report, Chapter 55 will be modified to remove the confusion.

CDFM Method

The conservative deterministic failure margin is an accepted approach to estimate HCLPF capacity (Reference 19A.2-6-1). The CDFM approach for calculating HCLPF values was approved by the USNRC for the AP600 plant. The same CDFM methodology, described in FSER 19A.2.2, is used for the AP1000 plant. The structural capacity is calculated based on concrete cracking and redistribution as described above under the heading Shear Modulus. ~~The values calculated for the shield wall, and documented in Chapter 55 of the AP1000 Probabilistic Risk Assessment report are:~~

~~Shield building Roof - Tension Ring~~	~~0.51g~~
~~Shield building Roof - at PCS Tank~~	~~0.63g~~
~~Shield building Roof - PCS Tank Wall~~	~~1.30g~~
~~Shield building Roof – Columns, Out of Plane Shear~~	~~0.74g~~
~~Shield building Roof – Columns, In Plane Shear~~	~~0.57g~~

~~As seen from above, the tension ring has the smallest capacity and consequently controls the shield building roof HCLPF value. Also, since tension controls the HCLPF value, shear strength does not.~~

In the calculation of the HCLPF values, inelastic energy absorption through ductility was not used concurrently with increased damping within the structural component being analyzed. ~~It is noted that the HCLPF values given are conservative since they do not reflect additional capacity due to damping and inelastic energy absorption. Therefore, t~~**Therefore there** is no double counting of damping values through the use of hysteretic damping for inelastic energy absorption and a damping value of 10 percent. ~~These margin factors were removed to address USNRC Open Items 720.449F Inelastic Energy Absorption, and 720.450F Damping (also discussed in FSER 19A.2.2.2 Inelastic Energy Absorption & FSER 19A.2.2.3 Damping).~~

~~Westinghouse's opinion is that the revisions as required by the NRC result in very conservative HCLPF values. Using damping and inelastic energy absorption as appropriate will result in realistic HCLPF values. Westinghouse is of the opinion that with the higher level earthquake, a damping factor of 1.1 can be used to represent the decrease in demand due to excessive cracking in the shield building. Further, a ductility factor 1.19 could be applied to the tension ring capacity that controls the HCLPF value. Therefore, a realistic HCLPF value for the shield building roof will be greater than 0.6g.~~

The shield building analyses are available for audit.

Uplift Effect of Foundation

See Westinghouse's response to DSER 19A.2-8.

References

19A.2-6-1 EPRI NP - 6041 - SL Rev. 1 August 1991: A Methodology for Assessment of Nuclear Power Plant Seismic Margin - Appendix L.

Design Control Document (DCD) Revision:

None

PRA Revision:

Modify the following sections under 5.5.2.3, Analysis of Structure Response, and Section 55.6, References.

Strength

This margin factor is defined from the finite element analysis based on the increase in seismic acceleration to failure based on ultimate stress criteria. ACI 349 provisions have been used to define ultimate strength for axial and flexure loads. For shear loads, the concrete and rebar capacities have been evaluated. **The total section shear strength is calculated with contributions of concrete and reinforcement together. The total in-plane shear strength including the contributions of the concrete and reinforcement ~~rebars contributions~~ uses Reference 55-12, Appendix L, "Shear Strength of Concrete Walls". The total out-of-plane shear ~~concrete~~ strength is based on beam action behavior taking into account the axial compression.**~~,~~
~~If the design shear load is greater than the concrete shear strength, the shear modulus has been increased to account for the shear strength in the reinforcement.~~

55.6 References

Add Reference 55-12

55-12 "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Electric Power Research Institute, EPRI NP-6041, Revision 1, August 1991.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-7 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

The applicant determined the HCLPF values on the basis of the estimated lower bound of qualification test results. When natural frequencies were not known, it was assumed that the equipment natural frequency coincides with the response spectra peak. When equipment frequencies are known and used for comparing the required response spectra (RRS) to the test response spectra (TRS), this information is to be included in the design specification. The applicant has not identified any equipment for which such design specification will be included. Although the applicant appears to have used a conservative approach to obtain the equipment HCLPF value from test results, it is not clear how the use of known natural frequency values for equipment within the standard design scope will be implemented. Since there are many electrical components with HCLPF values at 0.54g and one at 0.53g, electrical components may become critical in determining the Plant HCLPF value. This is Open Item 19A.2-7.

NRC Follow-On Comment:

Replace "will be" by "are" in the last sentence of first paragraph of Westinghouse response.

Westinghouse Response (Revision 1):

The design specification is part of the procurement package. The requirements to which the equipment is to be purchased are included in the design specification. This includes all those pieces or classes of equipment that have known frequencies that are used to define the HCLPF by comparing the RRS and TRS. These frequencies ~~will be~~are included in the design specification for the equipment to assure that the dynamic characteristics are the same as those expected.

Electrical components for non-safety systems are not critical in determining the plant HCLPF value since all SMA sequences are evaluated with loss of offsite power and loss of onsite AC power leading to a station blackout event. With the loss of power it has been shown that the plant design is robust against seismic event sequences each of which contain station blackout coupled with other seismic or random failures.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-9 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Generic Fragility Data

When HCLPF values could not be determined by using one of the methods described above, Westinghouse used generic fragility data. The cases where this approach was used are the following:

- Reactor internals and core assembly that includes fuel
- Control rod drive mechanism (CRDM) and hydraulic drive units
- Reactor coolant pump
- Accumulator tank
- Piping
- Cable trays
- Valves
- Main control room operation and switch stations
- Ceramic insulators
- Battery racks

The generic fragility data came from the Utility Requirements Document which was reviewed by the NRC. Therefore, the use of generic fragility data developed by a joint industry group in the Utility Requirements Document is acceptable. However, the applicant has not indicated what amplification factor, if any, was used to adjust the generic fragility data for the AP1000 configuration. The PCS water flow transmitter, located at Elevation 261' with a HCLPF value of 0.53 g, is likely to have an amplified seismic response. The applicant needs to justify the HCLPF values in the range of 0.53 g and 0.73 g that were obtained from the generic data as shown in the AP1000 PRA Table 55-1, Sheet 3 of 4. This is Open Item 19A.2-9.

NRC Follow-On Comment:

For the generic data used, compare the seismic demand with capacity of component.

Westinghouse Response (Revision 1):

~~The AP600 methodology used for generic fragility data is also utilized for the AP1000 plant.~~ No amplification factor was used to adjust generic fragility data for the AP1000 plant. These generic fragility data are considered representative of the anticipated capacity. ~~The fragility data is from ALWR URD, Volume III, ALWR Passive Plant, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules, Revisions 5 & 6.~~ **The Utility Requirements Document data (Reference 19A.2-9-1) are based on plant sites geographically distributed across the**

central and eastern United States. The generic data can be considered to provide a measure of the spatial variation of the seismic hazard east of the Rocky Mountains. Rock and four different soil types (EPRI Soil Categories 2 to 5) are considered. Therefore, the seismic input for these generic sites is considered representative to the AP1000 plant whose SSE demand is based on a 0.3g seismic level with a modified Reg. Guide 1.60 spectrum (increased in the higher frequency region around 25 hertz) located on a rock site. The layout and design of the AP1000 plant is similar to the generic plants below the operating deck (El. 135') where the bulk of the safety related equipment is located, with very little outside of the containment. Therefore, the seismic demand defined by frequency content and seismic level will be similar to those associated with the Reference 19A.2-9-1 defined generic plants that are located east of the Rocky Mountains.

Where the AP1000 will potentially have higher response is in the area of the PCS valve room and the ADS valves located in the containment interior structure at El. 169'. The safety equipment in these areas are the valves, piping, and cable trays. It is noted that the HCLPF value of the PCS water flow transmitter, identified in the issue summary located at Elevation 261', is based on actual test data and AP1000 defined seismic response and not the Reference 19A.2-9-1 generic data. A comparison is made for the valves using the median capacity expressed in terms of spectral acceleration as given in the Utility Requirements Document (Reference 19A.2-9-1) to the maximum AP1000 safe shut down earthquake (SSE) seismic response of the shield building at elevation 261', and the containment interior structure at elevation 169' in the anticipated frequency range of response (> 20 hz) for the valves in Table 19A.2-9-1. It is seen that there is satisfactory margin:

Table 19A.2-9-1 – Comparison of Generic Seismic Response and AP1000 SSE Seismic Response

Description	Median Capacity [Spectral Acceleration]	Ratio of Spectral Acceleration
Valves [Reference 19A.2-9-1]	9.0g	
AP1000 CIS Seismic Response ≤ El 169'	≤ 3.5g	≥ 2.6
AP1000 ASB Seismic Response ≤ El 334'	≤ 2.5g	≥ 3.6

The failure mode for piping and cable trays, as identified in Reference 19A.2-9-1, are the supports. These piping and cable tray systems for the AP1000 plant are designed for the peak response, or away from the peak, meeting code stress limits. Therefore, since piping and cable tray supports in general have significant reserve strength, it is anticipated that the HCLPF values will be greater than or equal to that calculated using the generic data.

~~In the AP1000 PRA~~**It is noted that in** chapter 55, Section 55.2.1, Westinghouse identified the following COL actions to confirm the seismic margin evaluation that includes generic fragility data:

> As part of a COL action, a qualification seismic review of the design will be performed with the purpose of identifying vulnerabilities and confirming the basis of the seismic margin evaluation. For each plant, a verification walkdown will be performed with the purpose of identifying differences in the as built from design and ensuring vulnerabilities were not created.

<u>References</u>

19A.2-9-1 ALWR URD, Volume III, ALWR Passive Plant, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules, Revisions 5 & 6.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

DSER Open Item Number: 19A.3-1 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Major SMA Model Assumptions

The applicant has used a PRA based seismic margin analysis method similar to the AP600 plant. In conducting its SMA, the applicant made the following assumptions:
- Seismic events occur at full power
- The review level earthquake (RLE) is 05 g
- The loss of offsite power occurs at the RLE. No credit is taken for non-safety related diesel generators for on-site AC power
- No credit is taken for non-safety related systems
- Initiating seismic event categories are derived from the AP600 model and the min-max method was used to calculate the plant HCLPF value

The staff notes that the seismic response of the AP1000 structures and some primary system components could be higher than those in AP600, because the height of the containment and the overall mass of API000 plant have increased. As indicated in the previous section of this report, it will be necessary to resolve the open items prior to the acceptance of the validity of plant seismic event trees derived from the AP600 model. This is Open Item 19A.3-1.

Original Westinghouse Response:

The seismic margin data (HCLPF) are based on AP1000 specific design characteristics and seismic response, and therefore reflect the differences in AP600 and AP1000 design.

The methodology used for the PRA and development of seismic fragility data (HCLPF values) remains unchanged between the AP600 and AP1000 plants.

Therefore, based on the above, the plant seismic event trees for the AP1000 are valid.

Additional NRC Comment:

The seismic responses in different areas of the AP1000 plant are different from the AP600 plant. This may affect the initiating seismic event categories, and boolean diagrams, event trees, etc. Westinghouse is to address this issue, and not the effect on HCLPF values.

Westinghouse Response to Additional NRC Comment:

AP1000 Seismic Margins Evaluation is presented in Section 55 of the AP1000 PRA Report. The initiating event HCLPFs are calculated in Section 55.4.2, referring to Tables 55-3 through 55-7 where the details of the calculations are shown. AP1000-specific SSC HCLPFs are used for these calculations. It is noted that:

1. Some of the AP1000 structure HCLPFs that could determine the plant HCLPF are lower than those of AP600 due to changes in the size of containment, tanks, etc. This is factored into the AP1000 plant HCLPF through the EQ-IEV-STRUC initiating event. One significant change is the reduction from 0.6g to 0.5g of the IRWST HCLPF. This affects both the EQ-IEV-STRUC and EQ-IEV-LOSP event trees.

2. The functional success criteria for plant response to initiating events have not changed from AP600 to AP1000 design. Thus, the structure of the event trees for determination of core damage sequences with only safety-related components credited does not change.

3. The plant HCLPF value is calculated to be the lowest HCLPF value of the SSCs that are credited for causing or mitigating a potential seismic core damage; namely 0.5g. Note that the failure of the ceramic insulators is postulated at a much lower g value. Thus, at least a transient event with loss of offsite power occurs.

4. During the AP600 SMA, a very interesting property of plant HCLPF calculations (using the min-max method) was observed. Namely, the following holds:

The plant HCLPF value is always greater than or equal to (but never is less than) the minimum of the initiating event HCLPF values, regardless of the HCLPF values of the mitigating systems modeled in the event trees. This enables us to calculate the plant HCLPF, without going through the HCLPF calculations for mitigating systems (event tree nodes) in the event trees, for all but the transient event (EQ-IEV-LOSP).

This property can be seen to hold as follows:

1. Every seismic CDF cutset contains an initiating event term. Thus, the HCLPF value for a cutset can not be lower than that of the initiating event HCLPF.

2. This can be visually seen as follows. Refer to Figure 1. In this figure, one of the two mitigating systems that prevent core damage has a smaller HCLPF than that of the initiating event, the other has a larger HCLPF value. Then, the event tree HCLPF value is equal to the initiating event HCLPF value.

If both system HCLPF values were smaller that that of the initiating event HCLPF value, then the event tree HCLPF would have been the same as the initiating event HCLPF.

Finally, if both system HCLPF values were greater than that of the initiating event HCLPF value, then the event tree HCLPF value would have been equal to the smaller of the two system HCLPFs, still being larger than the initiating event HCLPF.

Note that this example can be generalized to many more event tree nodes and sequences, still having the property mentioned above to be valid.

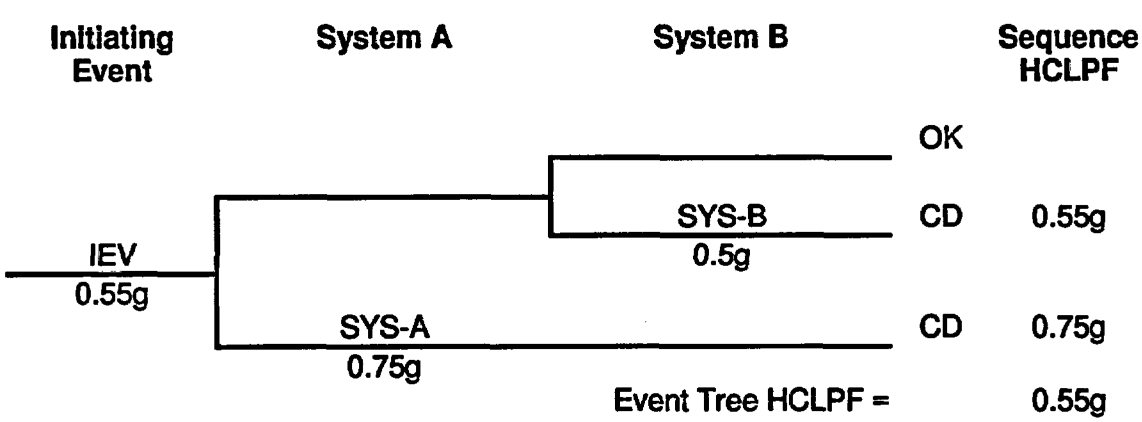
For the transient event EQ-IEV-LOSP, there are at least two success paths to avoid core damage (as modeled in the AP600 PRA seismic event tree):

1. Passive RHR successful. The PRHR system HCLPF is determined by the EQ-IRWST-TANK term, which has a HCLPF value of 0.5g. Failure of IRWST by the seismic event is already accounted for in the EQ-IEV-STRUC initiating event. Other components and actuation of PRHR have higher HCLPF values and do not drive the system HCLPF. Thus, this path does not introduce any insight into the plant HCLPF calculation.

2. PRHR fails, but CMT and ADS, and IRWST-injection, and IRWST-recirculation are successful. Again, the IRWST tank is the limiting HCLPF for this sequence. Thus, it determines the minimum HCLPF value for the EQ-IEV-LOSP event.

This property enables us to calculate the plant HCLPF directly from the lowest of the initiating event HCLPFs as 0.5g, as explained in Section 55.4.2. Thus, the plant Boolean expressions are not further developed. The general conclusions and insights obtained in AP600 for the seismic margins evaluation hold for the AP1000.

Figure 1. Illustration of the SMA Event Tree HCLPF Property



Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.3-2 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Initiating-Event Category HCLPFs

For all seismic event categories, except for the ECI-LOSP category, the HCLPF values of various seismic initiating event groups exceed 0.5 g. Each category of HCLPF group is discussed further below:

EQ-STRUC Group: The lowest HCLPF value of the Nuclear Island (NI) structure that can influence the plant HCLPF value is .05 g, based on the values shown in Table 55-1 of the AP1000 PRA. The HCLPF values shown in Table 55-1 need to be validated through the resolution of Open Item 19A2-8 discussed in the previous section. The applicant has assumed that there is no detrimental effect from any seismic interaction between the NI and the adjacent turbine, annex, diesel generator and radwaste building structures. The applicant has stated, "this assumption needs to be verified by a plant walkdown when an API000 plant is built." However, there is no entry on the COL interface requirement about the plant walkdown in Table 1.8-2 of the DCD. There is an entry in Table 1.6-2 19.59.10-1, "As-Built SSC HCLPF Comparison to Seismic Margin Evaluation," The applicant needs to justify why a specific item on plant walkdown verification of seismic interaction between the NI and adjacent structures is not included in the COL interface requirement. This is Open Item I9A.3-2.

NRC Follow-On Comment:

Westinghouse is to show that Chapter 1 points to walkdown verification like that given in AP1000 PRA Chapter 55, 55.2.1:

> "As part of a COL action, a qualification seismic review of the design will be performed with the purpose of identifying vulnerabilities and confirming the basis of the seismic margin evaluation. For each plant, a verification walkdown will be performed with the purpose of identifying differences in the as built from design and ensuring vulnerabilities were not created."

Westinghouse Response (Revision 1):

The COL Action Items associated with the AP1000 PRA are described in DCD Chapter 19.59.10.5 and PRA Section 59.10.5. The COL items are also identified in DCD Table 1.8-2 (Item 19.59.10-1, As-Built SSC HCLPF Comparison to Seismic Margin Evaluation). These COL items include ~~reconciliation of the as-built AP1000 to the AP1000 PRA, Seismic Margin Analysis.~~**a walkdown to identify differences in the as built from design and to ensure**

vulnerabilities were not created. ~~The method that the COL applicant reconciles the as-built AP1000 may include a plant walk-down.~~

The Combined License information provided in the AP1000 DCD Chapter 19 and in the AP1000 PRA Report will be revised to include a verification walkdown.

Design Control Document (DCD) Revision:

~~None~~

From DCD Revision 6, page 19.59-28h:

19.59.10.5 Combined License Information

The Combined License applicant referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 seismic margins analysis. A **verification walkdown will be performed with the purpose of identifying differences between the as-built plant and the design. Any** ~~d~~Differences will be evaluated to determine if there is a significant adverse effect on the seismic margins analysis results. Spacial interactions are addressed by COL information item 3.7-3. Details of the process will be developed by the Combined License applicant.

The Combined License applicant referencing the AP1000 certified design should compare the as-built SSC HCLPFs to those assumed in the AP1000 seismic margin evaluation. Deviations from the HCLPF values or assumptions in the seismic margin evaluation should be evaluated to determine if vulnerabilities have been introduced.

PRA Revision:

~~None~~

From PRA Revision 3, page 59-37:

59.10.5 Combined License Information

The Combined License applicant referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 seismic margins analysis. A **verification walkdown will be performed with the purpose of identifying differences between the as-built plant and the design. Any**~~d~~Differences will be evaluated to determine if there is significant adverse effect on the seismic margins analysis results. Spacial interactions are addressed by COL information item 3.7-3. Details of the process will be developed by the Combined License applicant.

The Combined License applicant referencing the AP1000 certified design should compare the as-built SSC HCLPFs to those assumed in the AP1000 seismic margin evaluation. Deviations from the HCLPF

values or assumptions in the seismic margin evaluation should be evaluated to determine if vulnerabilities have been introduced.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.3-3 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

EQ-SLOCA: The applicant included a number of elements of seismic fragility in this group. These elements include, simultaneous failure of all small diameter instrument lines, steam generator tube rupture, and large steam line breaks. Steam generator tube rupture event considers up to 5 simultaneous tube ruptures. The EQ-SLOCA grouping appears reasonable. However it is not clear if the applicant considered degradation of steam generator tubes under the full service life of steam generators for developing the seismic fragility. The applicant should explain how service related degradation of steam generator tubes was considered in the development of the HCLPF value of this group. This is Open Item 19A.3-3.

NRC Follow-On Question:

When considering the steam generator tube rupture event, why were 5 simultaneous tube ruptures considered?

Westinghouse Response (Revision 1):

~~Degradation of steam generator tubes under the full service life of steam generators was not considered in the development of the steam generator seismic HCLPF values. The HCLPF value for the steam generators is dominated by the failure of the SG supports with a conservatively estimated HCLPF of 0.54g. In the AP1000 PRA SMA, the failure of steam generators is already identified as one of the contributors to SLOCA initiating event plant HCLPF.~~

~~The HCLPF value for the AP1000 SG tubes, whether potential degradation during their operational life is considered or not, will not be significant enough to lower the existing steam generator HCLPF value. Degradation of steam generator tubes for the AP1000 plant will not be significant since new design features are incorporated into the AP1000 steam generator that reduces degradation, such as:~~

- ~~Reduced wear due to tighter manufacturing tolerances and through the selection of materials.~~
- ~~Use of stainless steel tube support plates that eliminate denting and high cycle fatigue associated with carbon steel tube support plates.~~
- ~~Use of thermally treated alloy 690 tube materials using better manufacturing methods~~

~~Also, the tubes are to be inspected throughout the steam generator service life following the EPRI guidelines.~~

~~Thus, even if it were possible to estimate the SG tube HCLPF with operational degradation considerations included, the HCLPF value would be expected to be higher than that of the SG supports, which already dominate the SG HCLPF.~~

Five simultaneous tube ruptures were considered consistent with common PRA practices. Further, Westinghouse uses the upper limit of five simultaneous steam generator ruptures consistent with the Commission and staff position as set forth in SECY –93-087 under 19.II.R.1, Multiple Steam Generator Tube Ruptures:

> **"The Commission approves the staff's position to require that analysis of multiple steam generator tube ruptures (STGRs) involving two to five steam generator tubes be included in the application for design certification for the passive PWRs. "**

Design Control Document (DCD) Revision:

None

PRA Revision:

None