Westinghouse

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

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

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

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July 7, 2003

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

This letter transmits 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/NRC1602"

2. Westinghouse Non-Proprietary Responses to US Nuclear Regulatory Commission DSER Open Items dated July 7, 2003

D063

Attachment 1

"List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1602"

Attachment 1

Table 1 **"List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1602"**	
3.8.4.5-2	14.3.2-6
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Attachment 2

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

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.8.4.5-2

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

Summary of Issue:

During the course of its review of the Wall 7.3 design calculation, the staff noted that the applicant had previously identified and corrected an error in the equation used by INITEC to calculate the required positive reinforcement for a section subjected to both bending moment and axial load. The staff could not conclude during the audit that the corrected equation accurately calculates required positive reinforcement. Therefore, the applicant was requested to submit the derivation of the equation currently used to calculate the required reinforcement. The applicant was also requested to submit a sample verification calculation for the computer algorithm, and verify that the corrected equation has been utilized in all calculations. This is Open Item 3.8.4.5-2.

Westinghouse Response:

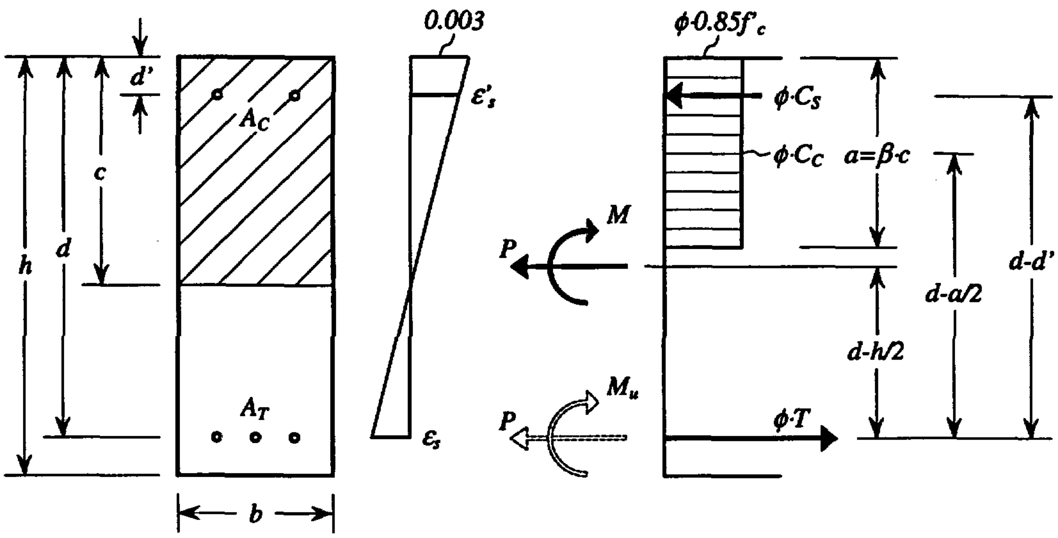
The development of the equation for sizing the required reinforcement for a section subject to bending moment and axial load is shown in this response. This equation is applicable when the strength of the section is controlled by yielding of the tension steel and both tension and compression steel, if any, are at yield. Loads and internal forces are shown in the following figures. The design loads P and M act at the centroid of the section, where:

P = Design Axial Load (P is positive in tension)
M = Design Moment

These loads are then converted to loads P and M_s relative to the plane of tension reinforcement.

The strength reduction factor ϕ is applied to both the steel and concrete strengths to obtain the design strength as shown in the figures.

Compression reinforcement is calculated such that the portion of tension reinforcement not equalized by compression reinforcement (A_T - A_C) does not exceed 75% of the A_b that would produce balanced strain conditions.



Transfer design loads P and M to the plane of tension reinforcement as P and M_u:

$$M_u \equiv M - P\left(d - \frac{h}{2}\right) \tag{1}$$

From equilibrium of forces and moments:

$$T = C_C + C_S + \frac{P}{\phi} \tag{2}$$

$$M_u = \phi \cdot C_C\left(d - \frac{a}{2}\right) + \phi \cdot C_S(d - d') \tag{3}$$

Where, the concrete stress block is defined in 10.2.7 of ACI 349 as follows:

$$C_C = b \cdot \beta \cdot c \cdot 0.85 f'_c \tag{4}$$

$$\beta = 0.85 \qquad (\because f'_c \le 4000 psi) \tag{5}$$

For tension controlled failure with both tension and compression reinforcement strain at or greater than yield:

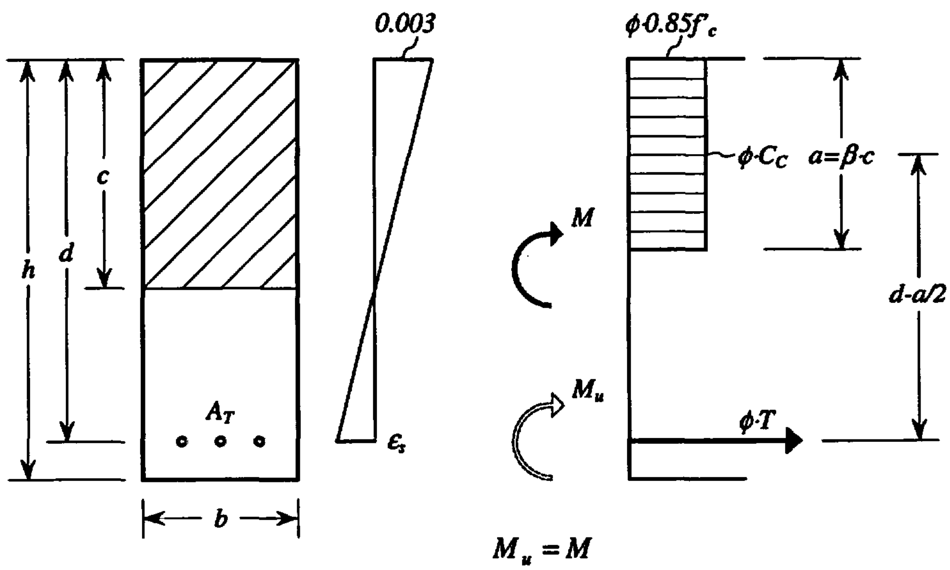
$$T = A_T \cdot f_y \tag{6}$$

$$C_S = A_C \cdot f_y \tag{7}$$

From equations (2), (6) and (7), the required reinforcement is given as follows:

$$A_T = \frac{C_C}{f_y} + A_C + \frac{P}{\phi \cdot f_y} \tag{8}$$

The third term on the right side can be calculated directly. The other terms are calculated to satisfy the limit of 75% of balanced strain conditions as described below:



From equilibrium of forces and moments:

$$T = C_C \tag{9}$$

$$M_u = \phi \cdot C_C \left(d - \frac{a}{2} \right) \tag{10}$$

From equations (4), (6) and (9):

$$A_T = \frac{b \cdot \beta \cdot c \cdot 0.85 f'_c}{f_y} \tag{11}$$

From equation (4):

$$a = \beta \cdot c = \frac{C_C}{b \cdot 0.85 f'_c} \tag{12}$$

From equations (6), (9), (10) and (12):

$$M_u = \phi \cdot C_C \left(d - \frac{C_C}{2 \cdot b \cdot 0.85 f'_c} \right) = \phi \cdot A_T \cdot f_y \left(d - \frac{A_T \cdot f_y}{2 \cdot b \cdot 0.85 f'_c} \right) \tag{13}$$

In the balanced strain condition, the neutral axis is calculated as follows:

$$\frac{c_b}{d} = \frac{0.003}{0.003 + f_y / E_s} = \frac{87000}{87000 + f_y} \tag{14}$$

Substituting the above c_b for c in equation (11), the balanced tensile steel A_b is given as follows:

$$A_b = \frac{b \cdot \beta \cdot 0.85 f'_c}{f_y} \cdot \frac{87000}{87000 + f_y} \cdot d \tag{15}$$

If M_u requires more reinforcement than 75% of A_b, compression reinforcement A_C is needed as 10.3.3 of ACI 349. Define M_{75} corresponding to 75% of balanced conditions as equation (13):

$$M_{75} \equiv \phi \cdot 0.75 A_b \cdot f_y \left(d - \frac{0.75 A_b \cdot f_y}{2 \cdot b \cdot 0.85 f'_c} \right) \tag{16}$$

The area of required reinforcement is calculated using the moment M_{75} as follows:

1) $M_u \le M_{75}$ (thus, compression reinforcement A_C is not required)

   Solving equation (13) for C_C:

$$\frac{C_C}{f_y} = \frac{0.85 f'_c}{f_y} \left\{ 1 - \sqrt{1 - \frac{2 \cdot M_u}{\phi \cdot b \cdot d^2 \cdot 0.85 f'_c}} \right\} \cdot b \cdot d \tag{17}$$

2) $M_u > M_{75}$ (thus, compression reinforcement A_C is required)

$$\frac{C_C}{f_y} = 0.75 A_b \tag{18}$$

$$M_u - M_{75} = \phi \cdot C_S (d - d') \tag{19}$$

From equation (7) and (19):

$$A_c = \frac{M_u - M_{75}}{\phi \cdot f_y (d - d')} \tag{20}$$

The equation has been verified against the following sample problems in the ACI Design Handbook (ACI 340.1R-91):

ACI Design Handbook		This Equation	Ratio
Flexure Example	A_S (in^2)	A_T (in^2)	A_T / A_S
3	1.11	1.10	0.99
10	17.7	18.0	1.02
17	1.84	1.84	1.00

- Flexure Example 3 – Determination of tension reinforcement area for rectangular beam subject to small axial load; no compression reinforcement
- Flexure Example 10 – Design of rectangular beam subject to simple bending; compression reinforcement found to be required
- Flexure Example 17 – Determination of tension reinforcement area for rectangular beam subject to bending and axial tensile load; top fiber found to be in compression

The corrected equation as developed herein has been used in all AP1000 calculations of reinforcement using the ANSYS post processors and EXCEL macros.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 10.2.8-1

Original RAI Number(s): 251.023, 251.024

Summary of Issue:

Westinghouse stated in Design Control Document (DCD) Tier 2 Section 10.2.3.1, "Materials Selection," that the turbine materials have the lowest fracture appearance transition temperature (FATT) and the highest C_V properties obtainable from water-quenched Ni-Cr-Mo-V material of the size and strength level used, indicating that suitable material toughness is obtained through the use of these types of material. Westinghouse's response to request for additional information (RAI) 251.023 resolved the NRC staff's concern about FATT and the nil ductility temperature (NDT). The response to RAI 251.024 dated March 25, 2003, clarified Westinghouse's fracture toughness requirements. This response indicates that the fracture toughness of the rotor materials will be at least 220 $MPa\sqrt{m}$ (200 $ksi\sqrt{in}$), and the ratio of fracture toughness to the maximum applied stress intensity factor for rotors at speeds from normal to design overspeed will be at least two. The staff finds these toughness and margin criteria to be acceptable because they are consistent with criteria approved for other applications involving assumed flaws, such as the reactor pressure vessel pressure-temperature limits. However, this criterion is not consistent with what is stated in DCD Tier 2 Section 10.2.3.4. This is DSER Open Item 10.2.8-1.

Westinghouse Response:

Please refer to the response to DSER-OI 10.2.8-3, which addresses this DSER-OI.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 10.2.8-3

Original RAI Number(s): None

Summary of Issue:

DCD Tier 2 Section 10.2.3 also states that the maximum tangential stress resulting from centrifugal forces does not exceed 65 percent of the 0.2 percent offset yield strength at design temperature and speed; or the tangential stresses will not cause a flaw, that is twice the corrected ultrasonic examination reportable size, to grow to critical size in the design life of the rotor. The first criterion is not consistent with the stress limit criterion of standard review plan (SRP) 10.2.3, which requires that the combined stresses of a low-pressure turbine disk at design overspeed due to centrifugal forces, interference fit, and thermal gradients not exceed 0.75 of the minimum specified yield strength of the material. This is DSER Open Item 10.2.8-3. The second criterion is not consistent with the margin of two between K_{Ic} and applied K mentioned earlier. This issue of inconsistency in the second design criterion is closely related to DSER Open Item 10.2.8-1 and will be resolved under it.

Westinghouse Response:

The turbine rotor design will meet the SRP 10.2.3 requirement that the combined stresses of a low-pressure turbine disk at design overspeed due to centrifugal forces, interference fit, and thermal gradients not exceed 0.75 of the minimum specified yield strength of the material. DCD 10.2.3.4 will be revised accordingly as identified in the Design Control Document (DCD) Revision portion of this document.

Although it is not necessarily obvious, the second criterion as presently stated in the DCD is not inconsistent with the preferred NRC statement that the ratio of fracture toughness to the maximum applied stress intensity factor for rotors at speeds from normal to design overspeed will be at least 2. However, should the initial flaw size be substantially larger than twice the reportable size, the ratio of fracture toughness to the maximum applied stress intensity factor for rotors at speeds from normal to design overspeed could be less than two. To satisfy the NRC concern, Westinghouse will revise the DCD as shown in the Design Control Document (DCD) Revision portion of this document.

Design Control Document (DCD) Revision:

Revise DCD 10.2.3.4 as follows:

10.2.3.4 Turbine Rotor Design

The turbine assembly is designed to withstand normal conditions and anticipated transients, including those resulting in turbine trip, without loss of structural integrity. The design of the turbine assembly meets the more restrictive of the following criteria:

- **The combined stresses of a low-pressure turbine disk at design overspeed due to centrifugal forces, interference fit, and thermal gradients do not exceed 0.75 of the minimum specified yield strength of the material**~~The maximum tangential stress resulting for centrifugal forces does not exceed 65 percent of the 0.2 percent offset yield strength at design temperature and speed~~; or,

- The tangential stresses will not cause a flaw, that is twice the corrected ultrasonic examination reportable size, to grow to critical size in the design life of the rotor. **This will result in the ratio of fracture toughness to the maximum applied stress intensity factor for the rotor at speeds from normal to design overspeed being at least 2.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 13.3-1

Original RAI Number(s): None

Summary of Issue:

13.3.3.3.4 TSC as a Vital Area
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 all 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.

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.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **13.3-2**

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

Summary of Issue:

<u>13.3.3.3.6 EOF as Alternate TSC</u>
Because of the unique design of the AP1000, the habitability system for the TSC is not the same as for the MCR. As such, the applicant states in DCD Tier 2 Section 18.8.3.5 that should habitability be challenged within the TSC, TSC personnel and functions are transferred to the EOF. This proposed arrangement is supported in DCD Tier 2 Section 13.3.1 with the COL information item proposing activation of the EOF when both onsite and offsite ac power is lost. In regard to TSC communications, DCD Tier 2 Section 1.8 states that communications systems and equipment outside the annex building (which includes the TSC) are site-specific elements and are outside the scope of the AP1000 standard plant, and that the DCD is based upon the COL applicant providing adequate external communications. The staff disagrees with this approach, in that the physical location of the EOF is not addressed, as it relates to the EOF serving as an alternate TSC. The distinction between transferring the TSC plant management function to the EOF upon loss of TSC habitability, rather than to the MCR (per section 2.6 of NUREG-0696), is also not discussed. Further, the condition of loss of both offsite power and onsite ac power to initiate EOF activation does not account for the second triggering event, in which high-high particulate or iodine radioactivity in the MCR air supply would also isolate the MCR from the TSC.

In the applicant's additional response to RAI 472.003, the use of the EOF as an alternate TSC is justified by the capabilities of the EOF, as well as when it is activated. In addition, the applicant states that the EOF design, including location, emergency planning and communications is the COL applicant's responsibility. TSC design requirements cannot be ignored based on unknown compensatory measures. If the EOF is the alternate TSC, its location will need to be evaluated against the following guidance criteria from Section 2.2 of NUREG-0696.

The onsite TSC is to provide facilities near the control room for detailed analyses of plant conditions during abnormal conditions or emergencies by trained and competent technical staff. During recent events at nuclear power plants, telephone communications between the facilities were ineffective in providing all of the necessary management interaction and technical information exchange. This demonstrates the need for face-to-face communications between TSC and control room personnel. To accomplish this, the TSC shall be as close as possible to the control room, preferably located within the same building. The walking time from the TSC to the control room shall not exceed 2 minutes. This close location will facilitate face-to-face interaction between control room personnel and the senior plant manager working in the TSC. This proximity also will provide access to information in the control room that is not available in the TSC data system.

The above discussion pertain to the TSC habitability and utilization of the EOF as an alternate TSC should be resolved. This is Open Item 13.3-2.

Westinghouse Response:

As stated in the response to DSER Open Item 13.3-1, Westinghouse believes that AP1000 meets all applicable requirements and guidance associated with providing TSC habitability.

Upon re-reviewing the regulations and guidance associated with the transfer of TSC functions in the event that the TSC becomes uninhabitable, Westinghouse will revise the DCD to be consistent with the guidance of NUREG-0696 section 2.6, Habitability. In that case, the TSC plant management function will be transferred to the main control room. The EOF will not be used as an alternate TSC.

Also, as the TSC personnel and functions are not going to be transferred to the EOF, the COL requirement to activate the EOF when both onsite and offsite ac power is lost will be removed from the DCD.

The AP1000 DCD will be revised as shown below in the, "Design Control Document (DCD) Revision" section of this Open Item.

Design Control Document (DCD) Revision:

Revise DCD 13.3 last paragraph as follows:

Staffing of the emergency operations facility occurs consistent with current operating practice and with revision 1 of NUREG-0654/FEMA-REP-1~~except for a loss of offsite power and loss of all onsite AC power. For this initiating condition, the Combined License applicant shall immediately activate the emergency operations facility rather than bringing it to a standby status~~.

Revise DCD 13.3.1 2nd paragraph as follows:

Combined License applicants referencing the AP1000 certified design will address the activation of the emergency operations facility consistent with current operating practice and NUREG-0654/FEMA-REP-1 ~~except for a loss of offsite power and loss of all onsite AC power. For this initiating condition, the Combined License applicant shall immediately activate the emergency operations facility rather than bringing it to a standby status~~.

Revise DCD 18.8.3.5, 7th paragraph as follows:

Should habitability be challenged within the TSC due to lack of cooling or a high radiation level resulting from a beyond-design-basis accident, the ~~TSC personnel and the functions~~**plant management function** of the TSC ~~are~~ **is** transferred to the ~~emergency operations facility (EOF) where habitability is not dependent on plant systems and with communication and data transfer links to the main control room to provide essential exchange of information~~**main control room**.

Revise DCD 18.8.3.5, 8th paragraph as follows:

~~A communicator is assigned to the main control room as part of the emergency staffing. The communicator is responsible for providing direct interface between the TSC and the main control room operators. If the TSC function has been transferred to the EOF, then the communicator provides the direct interface between the EOF and the control room operators.~~ The Combined License applicant is responsible for the EOF design, including the specification of its location (subsection 18.2.6) and emergency planning, and associated communication interfaces among the main control room, the TSC, and the EOF (Section 13.3).

Delete the 11th (i.e. next to last) paragraph of DCD 18.8.3.5

~~Providing an alternate source of power for the TSC functionality and habitability is not required. For those remote events that jeopardize the habitability of the TSC, transfer of the TSC functions and personnel to the EOF, provision of a communicator in the main control room, and assuring adequate communication and data transfer between the main control room and the EOF provide a reliable and flexible means of providing the TSC functions for those severe sequences that result in an uninhabitable TSC.~~

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-12

Original RAI Number(s): None

Summary of Issue:

Section 3.1, "Emergency Response Facilities," the staff finds this ITAAC unacceptable because it does not address the radiological habitability or the ventilation system for the technical support center; both of which should be the same as, or comparable to the main control room ITAAC. This is Open Item 14.3.2-12.

Westinghouse Response:

Westinghouse will revise the DCD as shown below.

Design Control Document (DCD) Revision:

DCD 3.1

- Add new item 6, under Design Description as follows:

  "6. The TSC provides a suitable workspace environment."

- Revise Table 3.1-1 to include new item 6 as follows:

6. The TSC provides a suitable workspace environment.	See Tier 1 Material, subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System	See Tier 1 Material, subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System

DCD 2.7-1

- Revise item 8.c), under Design Description and in Table 2.7.1-4 under Design Commitment as follows:

  The VBS maintains MCR and TSC habitability when radioactivity is detected.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-14

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:

As indicated in Section 3.8.2.1.1 of the Tier 2 portion of the AP1000 DCD, "The containment vessel is an ASME metal containment . . . Final detailed analysis will be documented in the ASME Design Report." Design Commitment 2.c) of ITAAC Table 3.3-6 in Section 3.3 of the Tier 1 portion of the AP1000 DCD states that: "The containment and its penetrations are designed and constructed to ASME Code Section III, Class MC." 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

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **14.3.3-17**

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.

Design Control Document (DCD) Revision:

<u>DCD Tier 1</u>

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 provide 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.**

DCD Tier 2

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 can provide up to 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 can provide up to 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 can provide up to 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

DSER Open Item Number: 14.3.3-18

Original RAI Number(s): None

Summary of Issue:

Table 3.2-1, item 10.i: Subsection 2.7.1 does not have an ITAAC related to RSR – there is nothing in the ITAAC that requires inspection, test, and analyses for the RSR and ventilation. Please clarify. This is Open Item 14.3.3-18.

Westinghouse Response:

DCD Tier 1 Subsection 2.7.1 will be revised as shown below to resolve the RSR discrepancy.

Design Control Document (DCD) Revision:

2.7.1 Nuclear Island Nonradioactive Ventilation System

Design Description

Revise item 8a as shown.

8. The VBS provides the following nonsafety-related functions:

   a) The VBS provides cooling to the MCR, TSC, RSR, and Class 1E electrical rooms.

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

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Revise item 8a as shown.		
8.a) The VBS provides cooling to the MCR, TSC, RSR, and Class 1E electrical rooms.	See item 12 in this table.	See item 12 in this table.

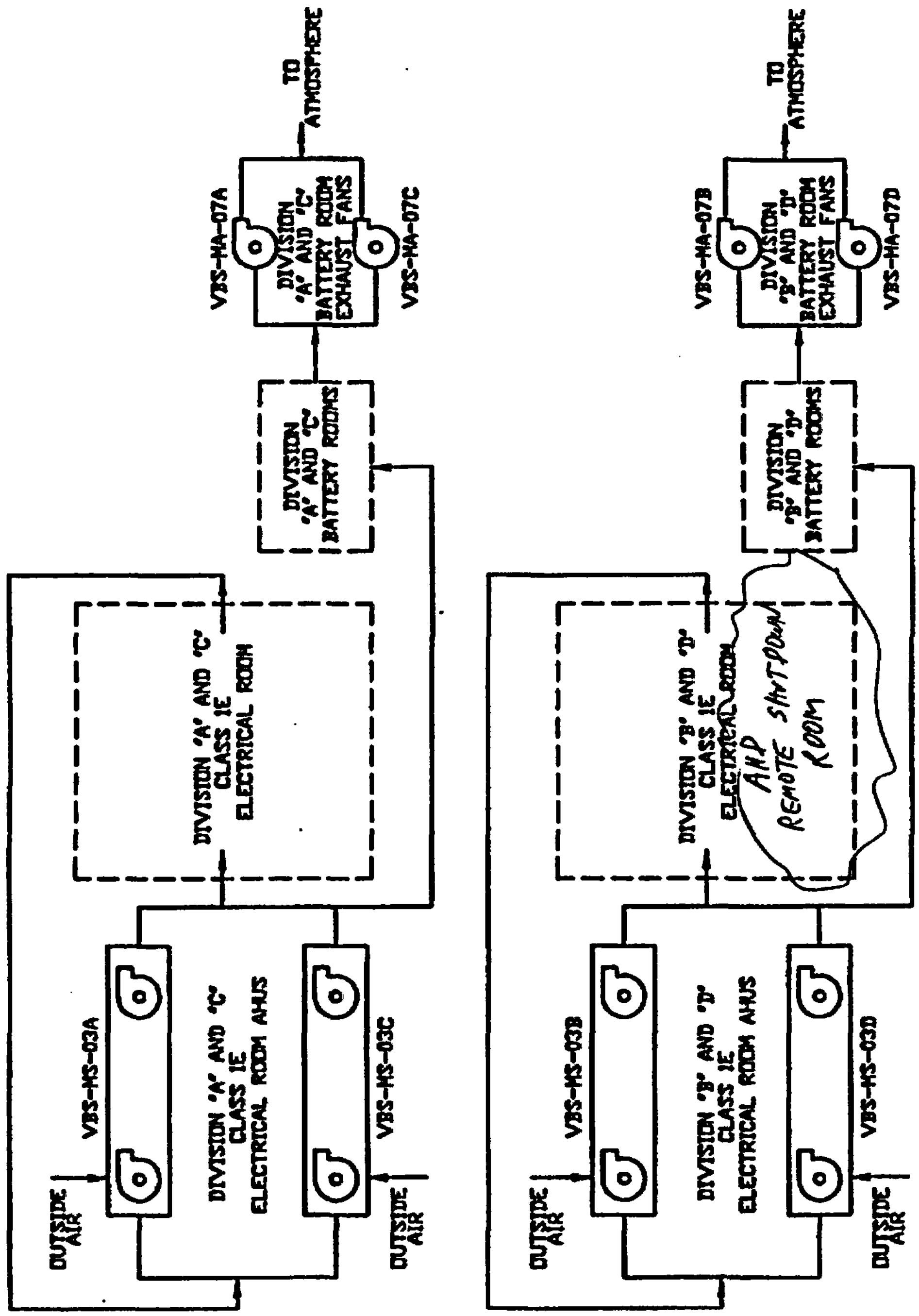


Figure 2.7.1-1 (Sheet 2 of 2)
Nuclear Island Nonradioactive Ventilation System

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 16.2-2

Original RAI Number(s): None

Summary of Issue:

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

Westinghouse Response:

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

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

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

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

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

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

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

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

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

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

AP1000 TS Comparison to the STS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

DSER OI 16.2-2, Table 1

Equivalent AP1000 and STS Technical Specifications

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

Allowable AP1000 TS Conditions

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

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

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

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

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

DSER Open Item 16.2-2, Table 2

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

Notes for Table 2

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

<u>Evaluation Summary</u>

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

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

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

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

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

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

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

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

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

Design Control Document (DCD) Revision:

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

LCO

This LCO establishes the minimum conditions necessary to ensure that sufficient accumulator flow will be available~~, assuming the minimum requirements of all other PXS components are met,~~ to meet the necessary acceptance criteria established for core cooling by 10 CFR 50.46 (Ref. 5). These conditions are:

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

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

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

LCO

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

DCD Chapter 16, TS 3.5.6, page 3.5.6-1

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

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

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

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	**One IRWST injection line actuation valve flow path inoperable.**~~motor operated containment recirculation isolation valve not fully open.~~ **OR** **One containment recirculation line actuation valve flow path inoperable.**	A.1	**Restore the inoperable actuation valve flow path to OPERABLE status.**~~Open motor operated containment recirculation isolation valve.~~	72 hours

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

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

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

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

If an **IRWST injection line actuation valve flow path or a containment recirculation line actuation valve flow path is inoperable,** ~~a motor operated containment sump isolation valve in one sump recirculation flow path is not fully open, the valve must be fully opened~~ **then the valve actuation flow path must be restored to OPERABLE status** within 72 hours. In this condition, three other **IRWST injection or containment** sump recirculation flow paths are available and can provide 100% of the required flow assuming a break in the direct vessel injection line associated with the other injection train, but with no single failure of the **actuation** ~~check~~ valve flow path in the same **injection** or sump recirculation flow path. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **19A.2-5**

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.

Westinghouse Response:

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.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 6.1.1-1

Original RAI Number(s): None

Summary of Issue:

The AP1000 DCD Tier 2 Section 6.1.1 indicates that DCD Tier 2 Section 6.2.5 contains the hydrogen production analysis for a post accident analysis. However, this statement is incorrect since the AP1000 DCD does not contain a hydrogen generation analysis in anticipation of NRC completion of a rule change that would eliminate the design-basis hydrogen accident. Since this is not consistent with the current rule, the staff is not able to complete a review of the corrosion rates and consequent hydrogen generation. Therefore, this is draft safety evaluation report (DSER) Open Item 6.1.1-1. Additional discussion related to this issue is contained in Section 6.2.5 of this report.

Westinghouse Response:

The AP1000 DCD contains the information consistent with the proposed NRC draft rule 10 CFR 50.44. The DCD will be revised as shown to remove the statement regarding hydrogen production analysis for a post accident analysis.

Design Control Document (DCD) Revision:

DCD Section 6.1.1.4 will be revised as follows:

In the post-accident environment, both aluminum and zinc surfaces in the containment are subject to chemical attack resulting in the production of hydrogen. The non-flooded surfaces would be wetted by condensing steam but they would not be subjected to the boric acid or trisodium phosphate solutions since there is no containment spray. ~~The hydrogen production analysis described in subsection 6.2.4 includes hydrogen generation due to corrosion processes and conservatively assumes that all surfaces are exposed to the solution.~~ **Nonsafety-related passive autocatalytic recombiners are provided to limit hydrogen buildup inside containment.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 6.2.5-1

Original RAI Number(s): None

Summary of Issue:

The NRC has proposed major changes to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," and related changes to 10 CFR 50.34 and 10 CFR 52.47, along with the creation of a new rule, 10 CFR 50.46a (see 67 FR 50374, August 2, 2002). These proposed changes are meant to risk-inform the combustible gas control requirements, and constitute significant relaxations of the requirements. The staff plans to finalize the rule changes during 2003. The AP1000 DCD is written in anticipation of these rule changes. As such, it is not in compliance with the current, more-restrictive regulations. Furthermore, until the proposed rule changes are final and effective, the staff cannot know for certain if the DCD will comply with the revised rule. Therefore, the issue of containment combustible gas control must remain open at this time. This is DSER Open Item 6.2.5-1.

Westinghouse Response:

The AP1000 DCD contains the information consistent with the proposed NRC draft rule 10 CFR 50.44.

Design Control Document (DCD) Revision:

See the Westinghouse response to DSER Open Item 6.1.1-1 that is related to this subject.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **14.3.2-6**

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

Summary of Issue:

Section 2.3.9, "Containment Hydrogen Control System," must remain open because hydrogen control is an open item in this report (See Section 6.2.5 of this report and Open Item 6.1.1-1 of this report for details). Briefly, this is because the AP1000 Tier 2 information is written in anticipation of a rule change to 10 CFR 50.44 that would relax requirements, but has not been finalized. This is Open Item 14.3.2-6.

Westinghouse Response:

The AP1000 DCD contains the information consistent with the proposed NRC draft rule 10 CFR 50.44. DCD Tier 1 Section 2.3.9, Containment Hydrogen Control System, is consistent with the comparable AP600 Tier 1 ITAAC. It has been modified to reflect the AP1000 design configuration.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **19.2.6-3**

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

Summary of Issue:

Overall Probability Distribution

The applicant has used the estimated median and COV values for the above failure modes, as shown in Tables 42-1 and 2 of the PRA for two sets of temperatures, 204 ° C (400 ° F) and 166 ° C (331 ° F), respectively. The applicant has developed the CCFP at the corresponding temperatures considering the above failure modes as independent. It is not clear whether or not the contribution to the CCFP from each equipment hatch has been taken as independent as well. Contributions to the CCFP from the equipment hatches are at least two orders of magnitude less than other contributions at 1.38 MPa (200 psig) internal pressure, and much less than that at lower internal pressures. Consequently, the influence of equipment hatch failure mode on the overall CCFP is negligible. Nevertheless, Chapter 42 of the PRA should be revised to clarify the approach used by the applicant. This is Open Item 19.2.6-3.

Westinghouse Response:

The contribution to the CCFP from each equipment hatch has been taken as independent. This statement is added to Section 42.4.3 for clarification.

Design Control Document (DCD) Revision:

None

PRA Revision:

Add the following statement to the end of Section 42.4.3 as a new paragraph (as the third paragraph):

The contribution to the CCFP from each equipment hatch has been taken as independent.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **9.5.2-3**

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

Summary of Issue:

The COL applicant should address the issue of Bulletin BL-80-15 for recommendations of loss of the emergency notification system due to a loss of offsite power. This is COL Action Item 9.5.2-3. Inclusion of this COL information in the DCD is Open Item 9.5.2-3.

Westinghouse Response:

DCD Tier 2 Subsection 9.5.2.5.1 will be revised as shown below to remind the COL applicant to review BL-80-15 for recommendations related to loss of the emergency notification system due to a loss of offsite power.

Design Control Document (DCD) Revision:

9.5.2.5.1 Offsite Interfaces

Combined License applicants referencing the AP1000 certified design will address interfaces to required offsite locations, **this will include addressing the recommendations of BL-80-15 (Reference 21) regarding loss of the emergency notification system due to a loss of offsite power.**

9.5.5 References

Add the following reference.

21. NRC Bulletin 80-15, "Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power," June 18, 1980.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 15.3.7-1

Original RAI Number(s): 630.052

Summary of Issue:

In RAI 630.052, the NRC staff asked the applicant for justification to not include a TS LCO for a decay time limit related to the assumption used in the radiological consequences analysis of the FHA. The applicant, in response to the RAI, stated that they had performed a sensitivity study in which the FHA was assumed to occur 24 hours after shutdown and the resulting doses remain below 10 CFR 50.34 guidelines. The applicant updated DCD to include a paragraph discussing an evaluation of the FHA performed assuming a decay time of 24 hours. The applicant asserts that a decay time LCO is not needed because the evaluation of the FHA at 24 hours shows that the capability of the AP1000 design to meet the regulatory dose acceptance criteria is not sensitive to the decay time. The staff's position is that in order to not include a technical specification LCO for the decay time, the design basis FHA dose analysis must assume a decay time that is clearly less than the time physically needed to begin moving fuel assemblies out of the core following unit shutdown for refueling. The staff does not consider 100 hours to be short enough. The applicant did not revise the design basis FHA dose analysis, which continues to include the 100-hour decay time assumption. Additionally, in DCD Tier 2 Section 15.7.4.5 discussing of the evaluation of the FHA at 24 hours, the applicant did not discuss the impact of the decay time on control room habitability due to an FHA at 24 hours. The staff does not consider this issue to be resolved. This is Open Item 15.3.7-1.

Westinghouse Response:

See the response to DSER Open Item 16.2-3. In that response, a new AP1000 TS 3.9.7, Decay Time, and the associated Bases were developed and will be included in the next revision to the DCD. The decay time TS LCO limit is 100 hours, which is the assumed radioactive decay time for the fuel handling accident in DCD 15.7.4.

Design Control Document (DCD) Revision:

None

PRA Revision:

None