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

Your ref: Docket No. 52-006
Our ref: DCP/NRC1598

June 23, 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.

We plan on submitting responses to all DSER Open Items by July 31, 2003, with a significant majority of our responses being provided by the end of June. In cases where our DSER response results in a revision to our Design Control Document (DCD) or other docketed report, our DSER response will include the specific wording change to the impacted document. We plan on submitting AP1000 DCD Revision 7, and other revisions to other docketed material, by August 29, 2003. These revisions will include the commitments provided in our DSER Open Item responses.

Westinghouse looks forward to working closely with the NRC in the near-term to develop a schedule to resolve DSER Open Items and therefore enable the NRC to issue a Final Safety Evaluation Report.

D063

Please contact me if you have questions regarding this transmittal.

Very truly yours,

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/NRC1598"
2. Westinghouse Non-Proprietary Responses to US Nuclear Regulatory Commission DSER Open Items dated June 23, 2003

DCP/NRC1598
Docket 52-006

June 23, 2003

Attachment 1

"List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1598"

Attachment 1

Table 1 "List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1598"			
2.5.1-1	3.8.2.1-1	6.2.1.8.2-1	14.3.3-1
2.5.2-1	3.8.2.2-1	6.2.1.8.3-1	14.3.3-2
2.5.4-1	3.8.3.5-1	6.2.1.8.3-2	14.3.3-3
2.5.4-2	3.8.3.5-2	6.2.1.8.3-3	14.3.3-4
2.5.4-3	3.8.3.5-3	6.2.6.4-1	14.3.3-5
	3.8.4.3-1		14.3.3-6
3.3.1-1	3.8.4.5-1	8.2..3.1-1	14.3.3-7
3.3.1-2	3.8.5.1-1		14.3.3-8
3.3.2-1	3.8.5.4-1	9.5.2-1	14.3.3-9
3.7.1.5-1	3.8.5.4-2	9.5.2-2	14.3.3-10
3.7.2.1-1	3.8.5.4-3	9.5.2-3	
3.7.2.3-2	3.8.5.5-1		16.2-1
3.7.2.3-3		14.2.7-1	
3.7.2.9-1	4.5.1-1	14.2.7-3	19.1.10.1-1
3.7.2.16-1	4.5.2-1	14.3.2-1	19.1.10.1-2
		14.3.2-2	19.4-1
	5.2.3-1	14.3.2-3	19A.3-1
		14.3.2-4	19A.3-3
		14.3.2-5	
		14.3.2-7	20.7-2
		14.3.2-9	
		14.3.2-10	
		14.3.2-11	
		14.3.2-13	

Attachment 2

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

DSER Open Item Number: 2.5.1-1

Original RAI Number(s): None

Summary of Issue:

The DCD Tier 2 information, while listing certain site specific aspects of basic geologic and seismic information to be provided by a COL applicant referencing the AP1000 certified design, does not include some of the attributes discussed above. This issue was discussed with the applicant during the April 2-5, 2003 audit. This is Open Item 2.5.1-1.

Westinghouse Response:

This Open Item was addressed by changes to Chapter 2 included in DCD Revision 5. The changes were made in the response to RAI 240.005 transmitted by letter DCP/NRC1586 on May 7, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 2.5.2-1

Original RAI Number(s): None

Summary of Issue:

COL applicants referencing the AP1000 certified design, without an ESP, should provide site specific information related to seismicity, geologic and tectonic characteristics of the site and region, correlation of earthquake activity with seismic sources, probabilistic seismic hazard analysis, controlling earthquakes, seismic wave transmission characteristics of the site, and the safe shutdown earthquake ground motion. The DCD Tier 2 information lists a number of these criteria; however, it should include probabilistic seismic hazard analysis, including the definition of controlling earthquakes. This issue was discussed with the applicant during the April 2 through 5, 2003 audit. This is Open Item 2.5.2-1.

Westinghouse Response:

This Open Item was addressed by changes to Chapter 2 included in DCD Revision 5. The changes were made in the response to RAI 240.005 transmitted by letter DCP/NRC1586 on May 7, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **2.5.4-1**

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

Summary of Issue:

The DCD describes the need for establishing a vertical face below the grade with lateral support of the adjoining undisturbed soil or rock and suggests the use of soil nailing to stabilize the vertical soil surface as an alternative method for achieving this provision. The stability of the nailed soil surface will depend on the length and depth of the soil anchors or nails. One result of this proposed construction technique is that the soil immediately surrounding the nuclear island (NI) consists of natural in-situ materials only, which have relatively continuous properties in the horizontal and vertical directions. Because this configuration conforms to the assumptions made in the seismic analyses performed to assess the seismic responses of the NI structures, the proposed excavation method is considered acceptable to the NRC staff. However, during discussions with the applicant during the November 2002 meeting, it was noted that the COL applicant should also show that the existing in-situ soil satisfies the minimum conditions (in terms of soil parameters) assumed for the design of the AP1000 foundation and exterior walls. In addition, if the in-situ soils are not appropriate for the use of soil nailing excavation techniques, the COL applicant should show that any other construction method planned for the excavation satisfies the assumptions of the design of the NI. If any other construction technique that requires excavation and backfill of large areas surrounding the NI is proposed, the procedures and criteria for installing the backfill should also be submitted by the COL applicants for review and approval. In addition, an evaluation of the effect of any alternative construction procedures on the seismic responses of the NI structures should be performed. The amount of lateral passive pressure used in the design of the NI needs to be specified as an interface requirement for the COL applicant. This issue was discussed with the applicant during the April 2-5, 2003, audit. This is Open Item 2.5.4-1.

Westinghouse Response:

This Open Item was addressed by changes to Chapter 2 included in DCD Revision 5. The changes were made in the response to RAI 240.005 transmitted by letter DCP/NRC1586 on May 7, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 2.5.4-2

Original RAI Number(s): None

Summary of Issue:

The bearing capacity of the subgrade is a fundamental design parameter for this standard design. In the design of the foundation of a large structure it is important to ensure that under normal operating conditions, the average pressure on the subgrade is less than the allowable average bearing capacity of the foundation material, and that the peak subgrade pressure caused by the load combination with the SSE imposing the largest toe pressure at the edge of the foundation is also within the allowable capacity of the subgrade. The allowable bearing capacity of the subgrade is governed by settlement or crushing. Under relatively soft soil conditions, short term soil movement due to water table fluctuation and long term settlement due to the super imposed loading affect the allowable bearing capacity. Under hard rock subgrade conditions, the bedding direction of rock layers and the level of cracking and other discontinuities in the matrix of the rock material can limit the allowable average and allowable peak bearing capacity. The response to the RAIs indicates that the bearing capacity at a hard rock site will exceed 21.55MPa (450,000 pounds per square ft). During the April 2 through 5, 2003 audit, the staff requested the applicant to clearly specify, in the DCD, that this standard design is based on an allowable average and an allowable peak bearing capacity, and should specify what these values are. This is Open Item 2.5.4-2.

Westinghouse Response:

This Open Item was addressed by changes to Chapter 2 included in DCD Revision 5. The changes were made in the response to RAI 240.005 transmitted by letter DCP/NRC1586 on May 7, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 2.5.4-3

Original RAI Number(s): None

Summary of Issue:

As stated in DCD Tier 2 Section 3.2, the nuclear island is the only seismic Category I structure in the AP1000 standard design. Differential settlement between the nuclear island foundation and the foundations of adjacent buildings does not have any adverse effect on the safety-related functions of structures, systems and components. Differential settlement under the nuclear island foundation could cause thebasemat and the building to tilt. In the narrow direction, the NI foundation width is 49.8 meters (163 feet and 6 in.) and the height above the bottom of the basemat is 83.3 meters (273 ft 3 in.). Assuming a basemat tilt of 10.2 centimeters (4 in.), the rigid body tilt at the highest point can be between 15 and 18 centimeters (6 and 7 in.). Under seismic excitation there will be an elastic deformation relative to the base. When these two effects are added, the annular space between the shield building and the containment structure will be diminished, and the functionality of the crane inside the containment and other sensitive components could also be affected. The DCD does not provide any quantitative justification as to why a basemat tilting of a few inches will not affect functionality of structures, systems and components. This issue was discussed with the applicant during the April 2-5, 2003, audit. This is Open Item 2.5.4-3.

Westinghouse Response:

Westinghouse does not consider tilting to be significant, and it will not impair the functionality of structures systems and components important to safety. The reasons for this position are the following:

- The nuclear island structures, consisting of the containment building, shield building, and auxiliary building are founded on a common reinforced concrete basemat foundation. Therefore, if tilting did occur due to settlement, all of the structures and equipment associated with the nuclear island would experience the same amount of tilt, and no interaction would occur. The space ("gap") between the shield building and the containment structures would not be diminished.

- The nuclear island structures are founded on a hard rock foundation. Therefore, any significant settlement is unlikely. The elastic deformations of the structures during a seismic event are included in the seismic analyses that have been performed.

- Differential settlement occurs primarily during construction before final alignment of equipment. Equipment could be readjusted, if necessary, for any subsequent long term tilting.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.3.1-1

Original RAI Number(s): None

Summary of Issue:

Pressure generated from the design wind velocity is further dependent on exposure and gust response factors corresponding to the exposure categories. The applicant has used exposure Category C which is consistent with open shoreline and flat open country exposure. Category C exposure is suitable for most sites in the Eastern United States; however, it is not suitable for sites near open inland waterways, the Great Lakes and coastal areas of California, Oregon, Washington and Alaska. The wind load design for AP1000 makes it unsuitable for sites that fall under the exposure Category D. Seismic Category I structures for AP1000 are robust and their lateral load resistance is generally governed by seismic and tornado loading. It may be feasible to demonstrate that the AP1000 wind design is adequate for exposure Category D. Without such a demonstration, the use of wind exposure category is an open issue. This issue is Open Item 3.3.1-1.

Westinghouse Response:

The basic wind speed of 145 mph selected for design is the maximum anywhere in the United States and occurs on the Eastern seaboard in hurricane prone areas in the Eastern United States. Exposure Category C is specified for design of the AP1000 in Section 3.3 because it is specifically identified in ASCE 7-98 as applicable to shoreline locations in hurricane prone areas.

Exposure Category D excludes shore lines in hurricane prone areas. It is applicable for sites near open inland waterways, the Great Lakes and coastal areas of California, Oregon, Washington and Alaska. For such locations the basic wind speed is less than the 145 mph used for design of the AP1000. Loads on the structures are based on the product of the square of the basic wind speed and coefficients based on the exposure category. At grade the velocity pressure exposure coefficients for exposure Category D are 21% greater than those for exposure Category C. At 200 feet above grade they are 10% higher. Thus, the AP1000 can be sited at sites with exposure Category D when the basic wind speed is equal to or less than 130 mph. The Combined License applicant will be able to demonstrate that the loads on the structure at these sites are lower than those used in design.

Design Control Document (DCD) Revision:

Revise Section 2.3 as follows:

The AP1000 is designed for air temperatures, humidity, precipitation, snow, wind, and tornado conditions as specified in Table 2-1. The Combined License applicant must provide information to demonstrate that the site parameters are within the limits specified for the standard design.

The design wind is specified as a basic wind speed of 145 mph with an annual probability of occurrence of 0.02. Wind loads are calculated for exposure C, which is applicable to shorelines in hurricane prone areas. The site parameters for the design wind may be demonstrated to be acceptable for other exposures or topographic factors by comparison of the wind loads on the structures. For example, for a site at a location with exposure Category D, the wind speed should be equal to or less than 130 mph.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.3.1-2

Original RAI Number(s): None

Summary of Issue:

In order to calculate the pressure loadings on structures for the design tornado wind velocity and the associated vertical distribution of wind pressures and gust factors, the applicant has used ASCE 7-98. The shape coefficients for the shield building, however, are calculated using American Society of Civil Engineers (ASCE) Paper No.3269, "Wind Forces on Structures," Vol. 126, Part II (1961). ASCE Paper 3269 is a reference in the Standard Review Plan in Section 3.3.1. It is not clear why the applicant used the latest ASCE standard for the basic wind velocity, importance category and exposure category, but did not use the recommendations of ASCE 7-98 for the velocity pressure and the corresponding pressure and force coefficients. AP1000 structures are dynamically rigid and the use of pressure coefficients different from those recommended by ASCE 7-98 is not likely to produce an unacceptable design, since the lateral strength of the AP1000 structures is likely to be governed by seismic and tornado loads. Nevertheless, the applicant should clarify its inconsistent use of the ASCE 7-98 recommendations for wind load design. This issue is Open Item 3.3.1-2.

Westinghouse Response:

The ASCE Paper 3269 is used for design of the shield building because it provides detailed shape coefficients for chimneys, tanks and similar structures. The simplified coefficients given in ASCE 7-98 do not specify the variation around the circumference. The detailed coefficients of the ASCE paper give total loads that are consistent with those specified in ASCE 7-98.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.3.2-1

Original RAI Number(s): None

Summary of Issue:

The procedures used to calculate pressure loads from the tornado wind velocity are the same as those used for wind, as discussed in Section 3.3.1 of this report. The tornado missile effects are determined using procedures discussed in DCD Tier 2 Section 3.5, and the acceptability of these procedures is given in Section 3.5 of this report. Tornado loading includes tornado wind pressure, internal pressure by tornado-created atmospheric pressure drop, and forces generated by the impact of tornado missiles. These loads are combined with other loads as described in DCD Tier 2 Section 3.8.4. The acceptability of these loads and load combinations is discussed in Section 3.8.4 of this report. The applicant has indicated that a maximum pressure drop of 13.8 kPa (2 psi) is used for non-vented structures, unless a lower value is justified by a detailed analysis using the provisions of ASCE 7-98 for partially vented structures. However, the applicant has not identified any structure within the scope of the AP1000 standard design for which a lower pressure drop has been used. Design certification is a final decision by the NRC subject to provisions of changes through rule making; consequently, the applicant needs to identify all the structures for which it has used a pressure drop lower than 13.8 kPa (2 psi). Therefore, the use of a tornado pressure drop of less than 13.8 kPa (2 psi) for vented structures in the future is an open issue. This issue is Open Item 3.3.2-1.

Westinghouse Response:

AP1000 nuclear island structures with the exception of the shield building have been designed as non-vented structures using a differential pressure equal to the maximum pressure drop of 2 psi. The portion of the shield building surrounding the upper annulus is designed as fully vented (zero differential pressure) due to the large area of the air inlets and discharge stack. Tornado loads on this portion of the shield building are only those due to the wind load.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.7.1.5-1

Original RAI Number(s): 241.001

Summary of Issue:

In Section 2.0, "Site Characteristics," and DCD Tier 2 Table 2-1, the applicant specified that the COL applicant will use the following design site-parameters to confirm the adequacy of the AP1000 seismic design for a specific site:

- The site-specific ground motion response spectra, defined at the foundation level, are bounded by the proposed design response spectra (the modified RG 1.60 ground response spectra) anchored to 0.3g as shown in DCD Tier 2 Figures 3.7.1-1 and 3.7.1-2.
- No potential for fault displacement is expected at the site.
- No liquefaction is expected at the site.
- The average allowable static bearing capacity is greater or equal to 402 kPa (8,400 psf) over the foot print of the NI at its excavation depth. The allowable bearing capacity under static plus dynamic loads exceeds 4,070 kPa (85,000 psf).
- The minimum shear wave velocity of the rock foundation is equal to or greater than 8,000 ft/sec.

Based on its review experience of other advanced reactors such as ABWR, System 80^{+} and AP600, the staff concludes that the above design site-parameters are reasonable and acceptable bounding limits for the COL applicant to use in confirming the adequacy of the AP1000 seismic design, except for the definition for the average allowable static bearing capacity for the hard rock site.

The staff requested the applicant to clarify whether this term refers to allowable strength or allowable displacement of the foundation. In its response to RAI 241.001, the applicant stated that the design will be acceptable for a hard rock site that has an allowable bearing capacity of 450 kips per square foot. The staff's review experience indicates that this is an extremely high value of "allow bearing capacity," that is difficult for the COL applicant to substantiate. Also, the response still did not clarify whether this definition refers to strength or displacement considerations. In addition, the review of the Civil/Structural Criteria document performed by the staff during the November 12 through 15, 2002, audit indicated that hard crystalline bedrock should have an allowable bearing capacity of four (4) kips per square foot. The definition of allowable bearing capacity for the hard rock site must also account for the influence of bedding direction, level of cracking and other discontinuities in the rock material which can serve to limit bearing capacity. These discrepancies need to be clarified by the applicant. The staff identified this as Open Item 3.7.1.5-1.

Westinghouse Response:

Additional information was provided in RAI 241.001 Response Revision 1 transmitted by Westinghouse letter DCP/NRC1557, dated March 26, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.7.2.1-1

Original RAI Number(s): 230.002

Summary of Issue:

In DCD Tier 2 Sections 2.5 and 3.7.1, the applicant proposed to found the NI structures on a hard rock site with an embedment of 39'-6". The staff's review identified a question regarding how lateral soil pressures due to embedment were calculated for use in the design of exterior walls of the NI. In its response to RAI 230.002 dated October 4, 2002, and January 21, 2003, the applicant stated that the exterior walls of the NI were designed for two lateral soil pressure cases: lateral earth pressure equal to the sum of static earth pressure plus the dynamic earth pressure, and lateral earth pressure equal to the passive earth pressure. The applicant also agreed to perform additional calculations of total earth pressures for the various load cases to ensure that the load case will lead to the maximum wall moments and shears. This is Open Item 3.7.2.1-1.

Westinghouse Response:

The response to RAI 230.002, Revision 1 revised the specification of lateral earth pressures and revised the DCD to remove any reference to 2D SASSI calculations. This revised specification and the resulting wall moments and shears were reviewed during the April 2-5 meeting. No additional calculations are required.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.7.2.3-2

Original RAI Number(s): 230.018

Summary of Issue:

In its response to RAI 230.18, the applicant provided the following justifications for these results which are summarized below:

(a) For the shield building roof structures, the maximum vertical absolute acceleration is 0.9 g for the AP600 and 0.89 g for the AP1000 (in the initial analyses). In the most recent AP1000 analyses, the dominating frequency is 5.81 Hz and the maximum absolute acceleration is 0.96 g in the vertical direction. These differences in seismic response are partly due to changes in modal properties but are also affected by the ground motion time history which envelops the design ground response spectrum.

(b) For the steel containment vessel, the maximum vertical absolute acceleration is 1.49 g for the AP600 and 1.40 g for the AP1000 (in the initial analyses). In the most recent AP1000 analyses, the dominating frequency is 16.97 Hz and the maximum absolute acceleration is 1.13 g in the vertical direction. The reduction in the vertical response is associated with better definition of the AP1000 polar crane and the use of a multi-mass model of the polar crane instead of the single mass model used in the AP600 analyses and the initial AP1000 analyses. The first frequency (representing the polar crane mode) of the combined model in the vertical direction is 6.415 Hz compared to that of 5.843 Hz in the previous analyses.

The applicant's justification provided in (a) above for the results of the shield building roof structures appears reasonable and is acceptable. As for the steel containment vessel (see (b) above), the staff's review of the DCD and the seismic analysis report of the steel containment vessel (Calculation APP-1000-S2C-037) during the November 12 through 15, 2002, audit, revealed that the first frequency (polar mode) of the combined model (combined vessel lumped-mass stick model with multi-mass polar crane model) is 6.415 Hz which is in the same range of that from the initial analyses. The frequencies (16.97 Hz and 28.201 Hz) and modal masses corresponding to the two dominating vertical modes of the revised steel containment vessel model (combined vessel model with multi-mass crane model) also remain essentially unchanged in comparison with those of the initial analyses. Because the frequency corresponding to the crane mode is more than 10 hertz apart from the dominating frequencies of the vessel, the staff does not expect the vertical absolute acceleration at the top of the AP1000 steel containment vessel would be significantly reduced due to the use of the multi-mass model of the polar crane. Based on the above discussion, the applicant needs to justify why the vertical acceleration at the containment vessel dome is reduced from 1.40g to 1.13g as a result of using different polar crane model. This is Open Item 3.7.2.3-2.

Westinghouse Response:

Additional information was provided in RAI 230.018 Response Revision 3 transmitted by Westinghouse letter DCP/NRC1588, dated May 13, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.7.2.3-3**

Original RAI Number(s): **230.020**

Summary of Issue:

The seismic model for the NI structures was developed on the basis of uncracked concrete section properties of shear walls. During the teleconference call on January 23, 2003, the staff questioned the applicant's assumption that the calculation of shear wall stiffness does not consider a reduction in stiffness due to cracking. The stiffness reduction would affect the seismic loads for the design of critical sections of the NI structures and the frequency locations of the floor response spectrum peaks. The applicant agreed to review the references provided by the staff on the stiffness reduction in shear walls, and provide justification or correction as needed. This is Open Item 3.7.2.3-3.

Westinghouse Response:

See response to DSER Open Item 3.7.2.3-1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.7.2.9-1

Original RAI Number(s): 230.020

Summary of Issue:

As described in DCD Tier 2 Section 3.7.2.9, the effects of parameter uncertainty had not been explicitly considered. To account for such effects, the applicant, following the guidelines of SRP Section 3.7.2 and RG 1.122, broadened the peaks of the floor spectra by ±15 percent based on the corresponding spectral peak frequency. The staff found this acceptable, except that Open Item 3.7.2.3-3 (see Subsection 3.7.2.3 of this report) concerning the issue of stiffness reduction due to shear wall concrete cracking remains to be resolved. This issue is especially significant when one considers the additional uncertainties associated with structural modeling. This is Open Item 3.7.2.9-1. This open item will be addressed in conjunction with Open Item 3.7.2.3-3.

Westinghouse Response:

See response to DSER Open Item 3.7.2.3-1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.7.2.16-1

Original RAI Number(s): None

Summary of Issue:

The seismic design basis earthquake for the AP1000 structures, systems, and components are essentially defined at the plant grade level in the free field by an SSE with the peak acceleration of 0.3g and the ground response spectra shown in DCD Tier 2 Figures 3.7.1-1 and 3.7.1-2. The seismic design of the NI features (structures including basemat, systems, and components) is predicated on the limitation of constructing the AP1000 at hard rock sites with shear wave velocity equal to 2438 m/sec (8,000 fps) or higher. If these design bases are not satisfied (i.e., the site condition is not within the range of site conditions specified in the DCD) or if the seismic analysis responses used for the design do not envelop the results obtained from a potential plant's site conditions other than the hard rock sites, the basis established for the design certification will no longer apply. The applicant should commit in the DCD (similar to the AP600 DCD) that the COL applicants should perform an analysis and an evaluation using the design basis earthquake ground motion and plant-specific site conditions to confirm the design adequacy of the AP1000 design. This is COL Action Item 3.7.2.16-1 and Open Item 3.7.2.16-1.

Westinghouse Response:

The DCD is being revised similar to the AP600 DCD so that the COL applicants may perform an analysis and an evaluation using the site specific earthquake ground motion and site conditions to confirm the design adequacy of the AP1000 design.

Design Control Document (DCD) Revision:

The following will be incorporated in the next revision of the DCD.

2.5.2.3 Sites With Geoscience Parameters Outside the Certified Design

If the site specific spectra at foundation level exceed the response spectra in Figures 3.7.1-1 and 3.7.1-2 at any frequency, or if soil conditions are outside the range evaluated for AP1000 design certification, a site specific evaluation can be performed. This evaluation will consist of a site-specific dynamic analysis and generation of in-structure response spectra to be compared with the floor response spectra of the certified design at 5 percent damping. The site design response spectra at the foundation level in the free-field given in Figures 3.7.1-1 and 3.7.1-2 were used to develop the floor response spectra. The site is acceptable for construction of the AP1000 if the floor response spectra from the site-specific evaluation do not exceed the AP1000 spectra for each of the locations identified below.

• Reactor vessel support	Figure 3.7.2-17, Sheets 1-3
• Containment operating floor	Figure 3.7.2-17, Sheets 4-6
• Coupled auxiliary and shield building at control room floor	Figure 3.7.2-15, Sheets 1-3
• Coupled auxiliary and shield building at fuel building roof	Figure 3.7.2-15, Sheets 4-6
• Coupled auxiliary and shield building at shield building roof	Figure 3.7.2-15, Sheets 13-15
• Steel containment vessel at polar crane support	Figure 3.7.2-16, Sheets 1-3

Site-specific soil structure interaction analyses must be performed by the Combined License applicant to demonstrate acceptability of sites that have seismic and soil characteristics outside of the site parameters in Table 2-1. These analyses would use the site specific soil conditions (including variation in soil properties in accordance with Standard Review Plan 3.7.2). The three components of the site specific ground motion time history must satisfy the enveloping criteria of Standard Review Plan 3.7.1 for the response spectrum for damping values of 2, 3, 4, 5 and 7 percent and the enveloping criterion for power spectral density function. Floor response spectra determined from the site specific analyses should be compared against the design basis of the AP1000 described above. These evaluations and comparisons will be provided and reviewed as part of the Combined License application.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.8.2.1-1**

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

Summary of Issue:

The containment vessel is an metal containment designed in accordance with the ASME Code. The information contained in this subsection is based on the design specification and preliminary

Westinghouse Response:

Westinghouse identified additional detailed analyses to be performed for the containment vessel in letter DCP/NRC1583, dated May 1, 2003. These analyses are available for NRC staff review and demonstrate that the AP1000 containment vessel satisfies the acceptance criteria documented in the DCD. design and analyses of the vessel. During an April 2-5, 2003 audit at Westinghouse, the applicant informed the staff that the final detailed analyses, to be documented in the ASME Design Report, are not available and will be the responsibility of the COL applicant. The staff expected that the final detailed analyses for the AP1000 steel containment would be submitted for staff review as part of the design certification process for AP1000. To complete the staff evaluation of the AP1000 steel containment design, the staff will need to audit the final detailedanalyses. This is Open Item 3.8.2.1-1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.8.2.2-2

Original RAI Number(s): 220.004

Summary of Issue:

Stability of the AP1000 containment vessel and appurtenances is evaluated using ASME Code, Case N-284-1, Metal Containment Shell Buckling Design Methods, Class MC, Section III, Division 1, as published in the 2001 Code Cases, 2001 Edition, July 1, 2001. Since the latest version of Code Case N-284-1 has not been endorsed by the staff, the staff, requested the applicant to "provide its technical justification for the acceptability of this code case by demonstrating an equivalent level of safety when compared to Code Case N-284, Revision 0 plus the supplemental requirements of AP600 DCD Appendix 3G". In its response to RAI 220.004 (Revision 0), the applicant confirmed that the AP1000 criteria are the same as the AP600 criteria previously accepted by the staff. Because the applicant has demonstrated that the criteria of code case N-284-1 are consistent with the staff position for the evaluation of the steel containment buckling documented in Appendix 3 G to the AP600 DCD, the staff finds this acceptable, except that Code Case N-284-1 has not been designated as Tier 2* material, for which any proposed change to these criteria will require NRC approval prior to implementation of the change. The staff notes that in the AP600 DCD, Appendix 3G is designated Tier 2*. This is Open Item 3.8.2.2-2.

Westinghouse Response:

Code Case N-284-1 will be designated as Tier 2* material.

Design Control Document (DCD) Revision:

3.8.2.2 Applicable Codes, Standards, and Specifications

[The containment vessel is designed and constructed according to the 2001 edition of the ASME Code, Section III, Subsection NE, Metal Containment, including the 2002 Addenda.] Stability of the containment vessel and appurtenances is evaluated using ASME Code, Case N-284-1, Metal Containment Shell Buckling Design Methods, Class MC, Section III, Division 1, as published in the 2001 Code Cases, 2001 Edition, July 1, 2001.]**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.8.3.5-1

Original RAI Number(s): 220.010

Summary of Issue:

In its response to RAI 220.010 (Revision 1), the applicant indicated that (1) responses to RAIs 230.006 and 230.007 related to DCD Tier 2 Section 3.7 provide the information to address the concern relating to seismic analysis methods and the techniques for combining spatial effects of three earthquake components used for the internal structures; (2) DCD Tier 2 Tables 3.7.2-1 to 3.7.2-7 provide numerical values for frequency and accelerations; and (3) adequate safety is maintained because code criteria stress limits are used.

The applicant has revised DCD Tier 2 Section 3.8.3.5 to replace "response spectrum" with "equivalent static"; and revised DCD Tier 2 Table 3.8.3-2 to clarify the models and methodology utilized for the various analyses of the structural modules. During the April 2 through 5, 2003, design audit, the staff reviewed Westinghouse Calculation Nos. APP-1000-S2C-034, Revision 1 (finite element model of the containment internal structures), APP-1100-S2C-002, Revision 1 (seismic equivalent static analysis for containment internal structures), and APP-1200-S2C-001, Revision 0 (finite element - seismic equivalent static analysis of the auxiliary shield building). These analyses show how the equivalent static analysis method was implemented for the containment internal structures and structures outside the containment. The calculations demonstrated that the maximum equivalent static accelerations obtained from the stick model time history analysis were used as input to the finite element models of the plant structures. To combine the structural responses due to the three components of earthquake motion, the calculations used either the SRSS method or 1.0, 0.4, and 0.4 method. Accidental torsion was also included in the two horizontal directions. Modal frequencies for the structures were determined and presented in the calculations. Seismic amplification for out-of-plane flexibility of walls and floors was accounted for in most cases by either including the flexibility in the seismic time history model or developing an amplification factor (such as the containment internal structure wall modules). One item that arose during the April 2 through 5, 2003, design audit, is that there is no technical guidance document that demonstrates how the flexibility of walls and floors other than critical sections will be considered in the seismic analyses. Based on the above discussion and the discussion in section 3.7.2 of this report, the staff finds that the seismic analysis method used by the applicant and the results obtained are acceptable, except for the concern regarding the lack of a documented method for considering out-of-plane wall and floor flexibility. The concern related to the method for considering out-of-plane wall and floor flexibility is Open Item 3.8.3.5-1.

Westinghouse Response:

The internal Westinghouse civil and seismic design criteria specify that flexibility of walls and floors must be considered in the seismic design. Guidance was provided to the structural designers by verbal discussion and electronic mail. Westinghouse will add written technical guidance in internal design documents. The following will be added to guide the designer.

Wall Flexibility and Amplification

The equivalent static acceleration analyses of the nuclear island consider the global seismic response by applying the acceleration at each elevation based on the maximum acceleration at the center of mass at each elevation from the time history analyses of the nuclear island stick model. The designer shall consider the seismic amplification for local out-of-plane flexibility of walls and floors. This can be done in the following ways:

- Demonstrate that the out-of-plane wall or floor slab dynamic response is rigid (fundamental frequency > 33 hertz) and therefore there is no additional amplification.
- Document that the out-of-plane flexibility of the wall or floor slab had been included in the model used to develop the seismic loading.
- Calculate fundamental frequency of out-of-plane wall or floor slab and amplify seismic loads based on applicable floor response spectrum.
- Use results from dynamic analysis of a more detailed model that includes the flexible wall or slab out-of-plane response (e.g. finite element shell dynamic model of the nuclear island as described in DCD subsection 3.7.2).

The designer must document that the structure is rigid or that flexibility has been accounted for in the seismic analysis in his design calculation.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.8.3.5-2**

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

Summary of Issue:

As described in DCD Tier 2 Section 3.8.3.5.7, a design summary report is prepared for containment internal structures documenting that the structures meet the acceptance criteria specified in DCD Tier 2 Section 3.8.3.5. During the April 2 through 5, 2003, audit, the applicant provided the preliminary Containment Internal Structures Summary Report for review. Since the Design Summary Report has not been completed, the staff could not perform its review of the report in accordance with SRP Section 3.8.3. As indicated in SRP Sections 3.8.3.I.4 and 3.8.3.II.4, the Design Report is reviewed and considered acceptable if it satisfies the guidelines of Appendix C to SRP Section 3.8.4. Based on the above discussion, review of the Design Summary Report is Open Item 3.8.3.5-2.

Westinghouse Response:

The preliminary Containment Internal Structures Summary Report that was provided for information during the audit in April has been completed. The design summaries of the critical sections of the structural modules were updated in Revision 5 of DCD subsection 3.8.3.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.8.3.5-3**

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

Summary of Issue:

Westinghouse calculation No. APP-1100-S2C-007, Revision 0, contains the design of the IRWST concrete-filled steel module walls. The staff reviewed the approach used to calculate the needed steel area of the structural walls. The calculation determined the necessary steel reinforcement area at various locations in each of the critical walls. This was done using the methodology contained in Westinghouse guidance document APP-GW-S1-008, Revision 0 (Design Guide for Reinforcement in Walls and Floor Slabs). During the audit, the applicant indicated that boundary elements are not needed for walls that frame into other walls since the other walls act as boundary elements. The staff found that the applicant's approach for the analysis and design does not meet the criteria of Chapter 21.6, "Structural Walls, Diaphrams and Trusses," of ACI-349-01 in which, the criteria for using boundary elements are specified. A similar issue is presented in Subsection 3.8.4.2 of this report under Open Item 3.8.4.2-1. This is Open Item 3.8.3.5-3.

Westinghouse Response:

The open item on boundary elements for reinforced concrete walls is addressed in the response to DSER Open Item 3.8.4.2-1. Inside containment the walls are constructed using concrete filled steel modules. Typical corner details for these modules are shown in DCD Figure 3.8.3-8, Sheet 1. The steel plates provide excellent confinement for the concrete and the stiffeners limit potential buckling. The thickness of walls is established by shielding and constructibility considerations and the stresses are low. The corner details and low stresses result in a design satisfying the intent of Chapter 21.6 of ACI-349-01.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.8.4.3-1**

Original RAI Number(s): **220.015**

Summary of Issue:

The applicant referenced DCD Tier 2 Figure 3D.5-9 in its response to RAI 220.015; however, it does not appear to have considered the effect of rapid increase in compartment temperature. Based on its review of the selected calculations, the staff could not reach a conclusion that the applicant has adequately addressed the effects of thermal transients on concrete filled steel modules. The analysis approach for the thermal transient inside the IRWST is discussed in Subsection 3.8.3.3 of this report, and has been found to be acceptable. However, for subcompartment locations inside containment (other than the IRWST) and locations outside containment, rapid heat-up of the steel plate of the structural wall modules must be considered in the analysis and design of the structural wall module. The concern is that for a rapid temperature transient, the mismatch in thermal conductivity between the steel faceplate and the concrete could impose significant thermal stresses on the faceplate, studs, and concrete core. This could potentially result in degradation of the faceplate/concrete bond and invalidate the assumption of composite behavior. The applicant needs to evaluate the thermal transients that can occur in the various subcompartments, and demonstrate that no unacceptable degradation would result from differential thermal expansion of the steel and concrete throughout the entire transient. This is Open Item 3.8.4.3-1.

Westinghouse Response:

The effects of temperature transients resulting from postulated breaks have been evaluated. Temperatures are used from the containment analyses described in DCD Subsection 6.2.1.1.3. These analyses provide both subcompartment atmospheric temperatures and through wall temperatures in the heat sinks based on one dimensional heat flow. The structures are evaluated for the through wall temperature profile at critical times during the transient.

This response evaluates the initial part of accident thermal transients inside containment and demonstrates that the shear studs, and concrete within the Containment Internal Structure (CIS) structural modules retain their structural integrity. The structural modules outside containment are not subject to rapid thermal transients.

Thermal Transient

Structural modules are subjected to a rapid temperature transient in the event of a loss of coolant accident (LOCA) or a main steam line break (MSLB). The steel plate of the structural modules heats up most rapidly in the LOCA event. DCD Figure 6.2.1.1-6 shows the containment temperature response for thedouble ended cold leg break. Figure 3.8.4.3-1-1 shows the temperature distributions in the surface zone of the 30" thick structural module wall during the initial part (2000 seconds) of this LOCA transient. The relative distance shown on this plot is the ratio of the depth to the full thickness of the wall so the ½" thick plate is a relative distance of 1/60 = 0.0167. These curves show that there are high differential thermal gradients. The initial conditions in this thermal analysis are assumed to be 120 °F. Temperatures as high as 270°F are obtained in the first few minutes. The heat up of the plate would be slower but reach similar magnitudes for initial temperatures of the wall of 50 °F. The faceplate of a structural module may see differential temperatures ranging from 140°F to 220°F (based on an ambient temperature of 50°F). The concrete heats up more slowly and does not see a significant temperature increase during this early part of the transient. This results in relative thermal expansion of the faceplate, causing shear loads in the shear studs and embedded angles of the structural steel trusses that are welded to the faceplate. It also causes cracking of the concrete once the forces in the steel plate exceed the tensile strength of the concrete wall.

Structural Module

The structural modules in the containment internal structures (CIS) are described in the Design Control Document, Section 3.8.3.1. Two materials are used for the faceplates: (1) A36; and (2) Nitronic 33, ASTM A240 Type XM-29 UNS designation S2400. Shear studs on A 36 faceplates are ¾-inch diameter, 6 inches long, spaced at 10 inches each way. Shear studs on Nitronic 33 plates are the same size, spaced 10 inches horizontally and 8 inches vertically. The faceplate thickness is ½ inch.

The thermal growth of the faceplate on the structural modules is primarily uniaxial due to the restraint from the adjacent walls, floors, and ceilings. The configuration of the structural modules is shown in DCD Figure 3.8.3-1. The surface plates of the rooms (see DCD Figures 1.2-6 and 1.2-7) formed by the structural modules are as follows:

- Refueling Cavity – The faceplates are stainless steel. The adjacent walls restrain them horizontally. They are free to grow upwards.
- IRWST – The faceplates are stainless steel. The faceplates are not subject to heat up during the initial stages of the LOCA transient since the IRWST is full of water.
- Steam Generator and Pressurizer Compartments – The faceplates are carbon steel. The adjacent walls restrain them horizontally. They are free to grow upwards.
- Maintenance floor and vertical access – The faceplates are carbon steel. The adjacent walls restrain the approximately 90 degree corners horizontally. Embedded steel box columns assist the shear studs in providing horizontal restraint at the approximately 270 degree corners. The operating or maintenance floors restrain them vertically.
- CVS room – The faceplates are carbon steel. The adjacent walls restrain them horizontally. The maintenance floor restrains them vertically.
- Reactor vessel cavity – The faceplates are stainless steel. The adjacent walls restrain them horizontally. They are free to grow upwards.

The thermal evaluation shows that the fully restrained surface plates yield in compression when the temperature differential between the surface plate and concrete exceeds about 170°F. In regions where the walls are able to expand, the restrained growth of the surface plate loads the shear studs, which load the concrete in tension. This tension is sufficient to crack the concrete thereby reducing the stress in the surface plate and the loads on the shear studs. The maximum loads on the shear connectors occur in edge regions of the plate at locations where the concrete is restrained such that the wall may not crack. This restraint of horizontal growth of the concrete in the structural module wall occurs close to the bottom of the structural module where the thick basemat is slow to heat up and does not crack. It may also occur at locations above and below the floors where there is additional horizontal restraint provided by the floors. This case is evaluated using a uniaxial strip model of the steel plate and studs with the assumption of full restraint for the concrete.

Uniaxial Strip Model

The uniaxial strip plate shown in Figure 3.8.4.3-1-2 represents a single row of shear studs in either the horizontal or vertical direction. A typical wall is considered with a length or height of 30'. Similar uniaxial models were evaluated for the AP600. The AP600 evaluation used a continuum representation of the shear studs to evaluate the interface between the steel plate, studs and concrete. The AP1000 evaluation uses a discrete representation of each shear stud.

The formulation for the AP1000 thermal evaluation is given in Figure 3.8.4.3-1-3. The load in the shear stud is determined from a non-linear shear stud load-deflection curve based on Ollgaard and Slutter 1971 tests (Reference 3.8.4.3-1-1) as shown in Figure 3.8.4.3-1-4. The shear stud can transmit a peak load (24 kips) to the concrete even if there is localized deformation and crushing of the concrete in the vicinity of the shear stud. The test results show continued load capacity at deflections above 0.10 inches. This deflection of 0.1" is chosen as a representative limit for the evaluation. The force in a shear stud (F_i) and its displacement is a function of the axial stiffness of the plate (AE/L_k), the net force in the plate at stud location i, and the movement of the plate caused by the forces in the shear studs below the stud at location i.

The force in the central portion of the plate (F_{PL}) is limited by its yield stress. The maximum load that the shear studs must restrain is that corresponding to yield of the plate. The most critical thermal case is the one associated with a faceplate temperature of 240°F for both the carbon steel and stainless steel plates. Lower faceplate temperatures will not stress the faceplate to yield; higher faceplate temperatures will reduce the yield stress and hence the axial load that the shear studs must resist.

The shear stud load and deflections calculated using the uniaxial model for the carbon steel plate are shown in Figure 3.8.4.3-1-5. Figure 3.8.4.3-1-6 shows the stress in the faceplate when the surface plate is heated to 240°F during the initial part of the thermal transient. The maximum stress reaches yield in the central region of the plate. The maximum deflection in the shear stud (near the edge of the plate) is 0.057" (0.075" for the stainless steel plate) and is less than the established maximum limit of 0.1". The effective load length of the shear studs on the carbon steel plate that act to restrain the plate is 7 ½'. This is consistent with the design criterion used for the stud spacing that required that the studs over a length of 7.5 feet be capable of developing yield in the plate.

Mechanical loads

Dead and live loads that could occur during the thermal transient are not significant. Loads due to the safe shutdown earthquake are also small. The combination of the safe shutdown earthquake and thermal transient would have an extremely low probability since it is an independent event occurring during the few minutes after the LOCA when the maximum difference in temperature occurs between the steel surface plate and the concrete. Mechanical loads on the studs would not occur in the direction of free growth. They may occur in the direction normal to the free growth. These loads are small relative to those due to the thermal growth in the other direction and the biaxial loading would be acceptable.

Conclusion

The heat up of the surface plates during the initial portion of the LOCA transient results in cracking of the concrete walls except in regions where there is significant external restraint. This cracking reduces the stresses in the surface plates and the loads on the shear studs relative to the cases where there is significant external restraint. The cracking of the concrete does not cause degradation of the structural integrity of the wall.

In regions where there is significant external restraint, the structural module faceplates are restrained so that their thermal growth is uniaxial. This evaluation, using the uniaxial model with no growth of the concrete, demonstrates that the design is acceptable for the AP1000 thermal transients. Portions of the plate away from a free edge will reach yield. There are no shear loads on the studs in this central portion. The shear studs on the portions of the plate near the edge do not exceed the maximum deflection capacity. Loads in the plate and studs will be lower if there is also thermal growth of the concrete or if there is cracking.

References

3.8.4.3-1-1 Ollgaard, Jorgen, Roger Slutter, John Fisher, "Shear Strength of Stud Connectors in Lightweight and Normal-Weight Concrete," AISC Engineering Journal, April 1971.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

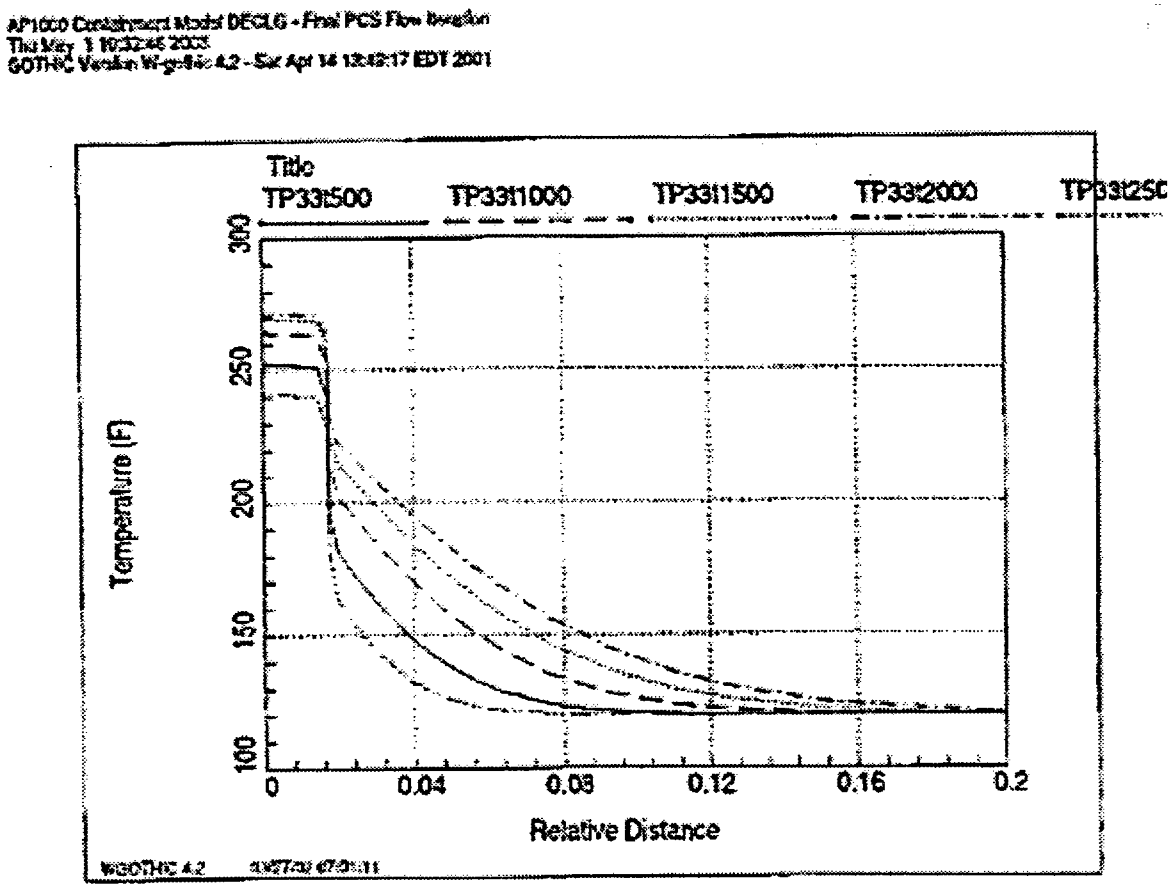


Figure 3.8.4.3-1- 1 – Temperature Profiles in Structural Module Wall During Initial Part of Thermal Transients

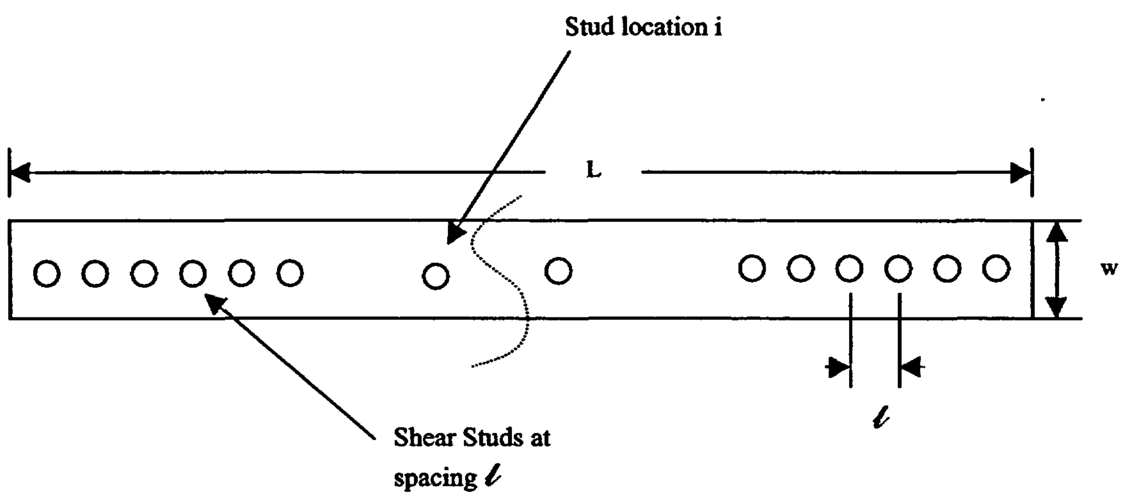
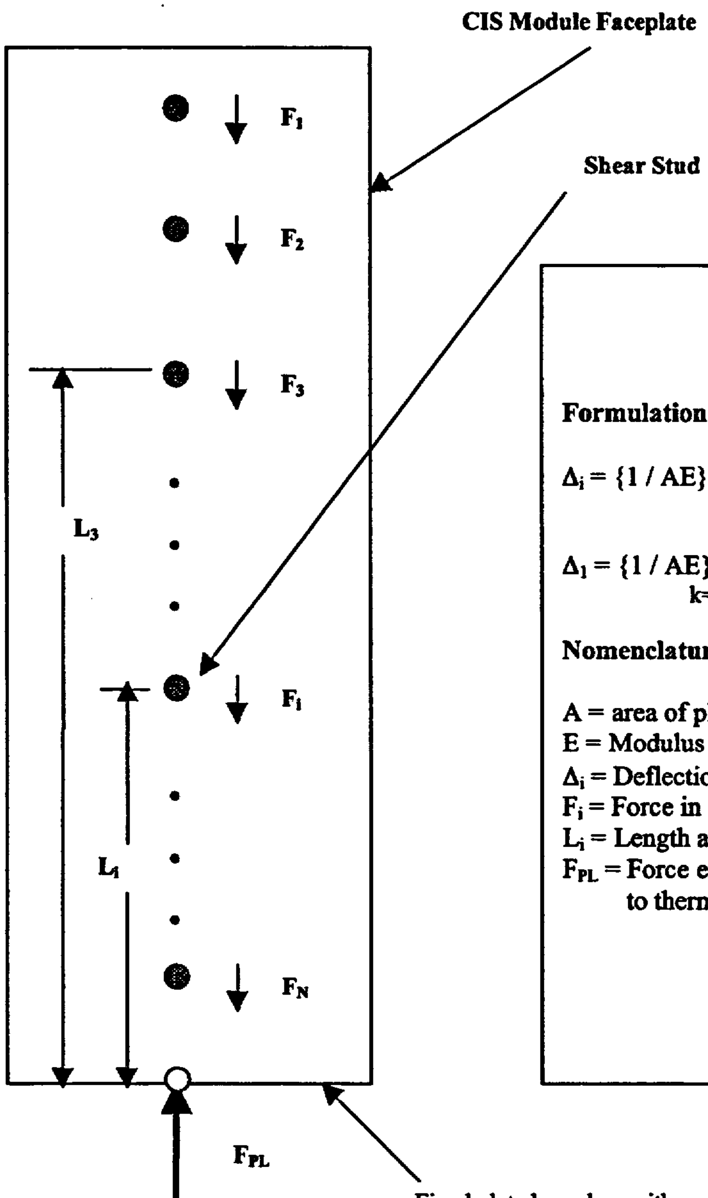


Figure 3.8.4.3-1- 2 – Uniaxial Strip Plate



Formulation

$$\Delta_i = \{1 / AE\} \left[\left(L_i \times \sum_{k=1,i} F_k\right) + \sum_{m=i+1,N} \left(F_m \times L_m\right) \right]$$

$$\Delta_1 = \{1 / AE\} \left[\sum_{k=1,N} \left(F_k \times L_k\right) \right]$$

Nomenclature

A = area of plate
E = Modulus of elasticity of plate
Δ_i = Deflection of shear stud i
F_i = Force in shear stud i; $F_i(\Delta_i)$, see Figure 220.015-4
L_i = Length as shown in figure
F_{PL} = Force exerted on shear studs from plate due to thermal event

F_N is defined so that the plate maximum stress is not exceeded (e.g., plate yield stress)

Figure 220.015-3 – Thermal Displacement

Figure 3.8.4.3-1- 3 – Thermal Displacement Mathematical Formulations

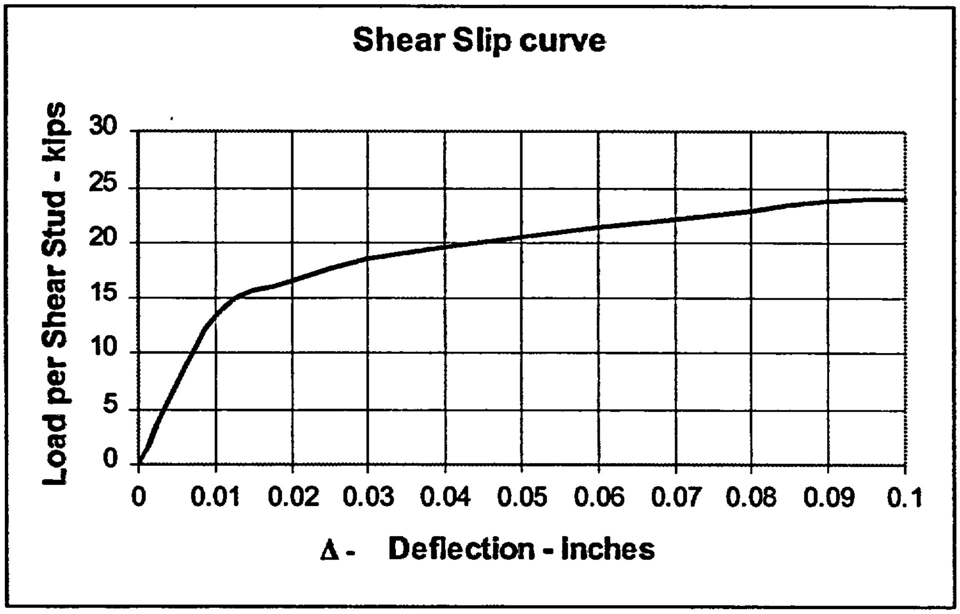


Figure 3.8.4.3-1- 4 – Shear Stud Load Deflection Curve

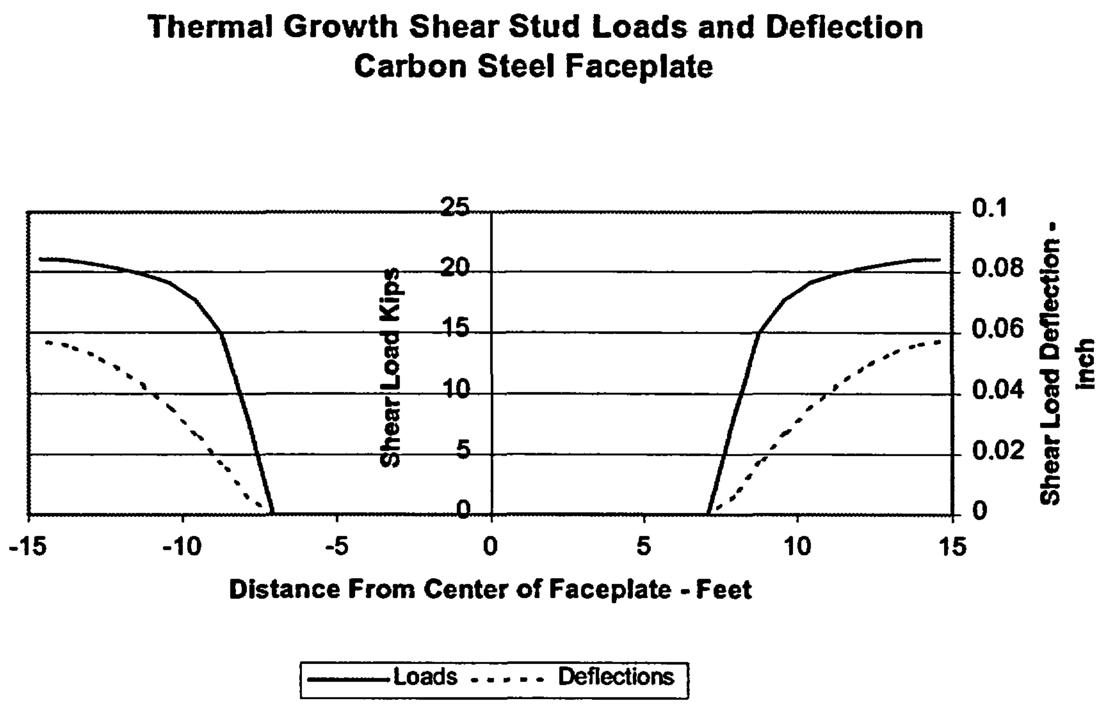


Figure 3.8.4.3-1- 5 – Thermal Growth Shear Stud Loads

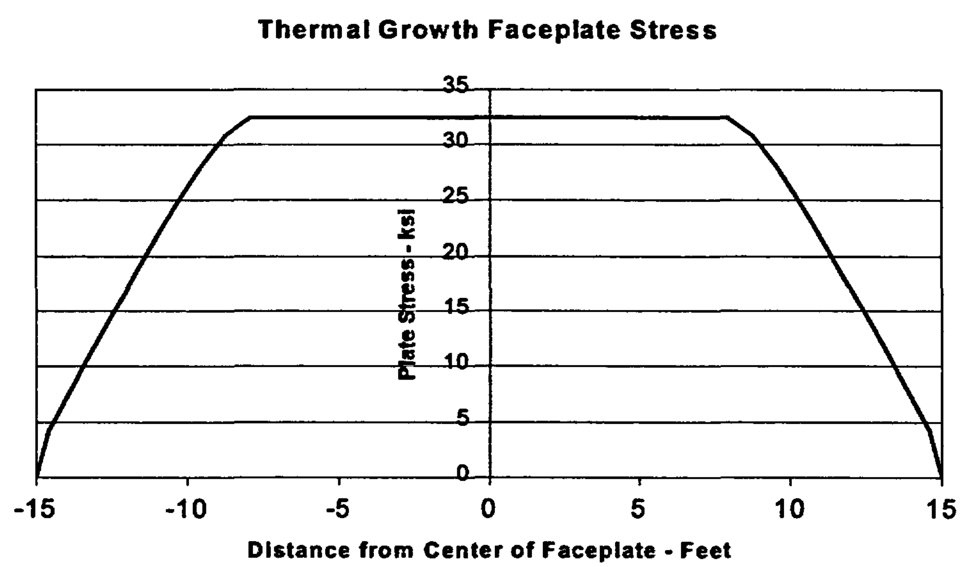


Figure 3.8.4.3-1- 6 – Thermal Growth Faceplate Stress

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.8.4.5-1**

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

Summary of Issue:

The staff notes that the final design of critical sections for other seismic Category I structures is incomplete at this time. This is addressed by Open Item 3.8.4.5-1 in Section 3.8.4.5 of this report.

Westinghouse Response:

The design summary report for the other Category I structures (auxiliary and shield building) is provided in DCD Appendix 3H. This appendix is updated in DCD Revision 6 to show the summary of the design of critical sections that were reviewed by NRC staff during the April 2 through 5, 2003, design audit.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.8.5.1-1**

Original RAI Number(s): **230.23**

Summary of Issue:

In DCD Tier 2 Section 3.8.5.1, the applicant states that the foundation is built on a mud mat, for ease of construction. The mud mat is lean, nonstructural concrete and rests upon the load-bearing rock. Waterproofing standards are described in DCD Tier 2 Section 3.4.1.1.1. In RAI 230.23, the staff raised a question that the non-structural concrete mud mat cannot withstand the very high toe pressure predicted in the applicant's seismic analysis. This may crush the non structural concrete mud mat and potentially affects the safety of the NI foundation mat under design basis combination of loads. Since the applicant did not provide a response to the RAI, this issue is designated as Open Item 3.8.5.1-1.

Westinghouse Response:

Additional information was provided in the RAI 230.023 Response transmitted by Westinghouse letter DCP/NRC1588, dated May 13, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.8.5.4-1**

Original RAI Number(s): **230.022**

Summary of Issue:

In its review of DCD Tier 2 Section 3.8.5.4, the staff determined that the potential uplift and sapping back of the containment internal structures foundation on the basemat through the steel containment vessel during a seismic event could affect both the seismic design loads and in structure response spectra for all structures, systems and components associated with the containment internal structure, and could also affect the seismic response of the steel containment shell. In RAI 220.021, the staff requested the applicant to perform additional analyses to demonstrate how the uplifting effect will be addressed, and how the uplifting effect on the seismic analysis results will be used for the design of the containment and containment internal structures. This is Open Item 3.8.5.4-1.

Westinghouse Response:

Additional information was provided in the RAI 230.022 Response transmitted by Westinghouse letter DCP/NRC1592, dated May 21, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.8.5.4-2**

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

Summary of Issue:

The staff notes that the specified shear wave velocity for a hard rock site condition should be 2438m/sec (8000 fps), instead of 1067m/sec (3500 fps). This has been previously discussed with and agreed to by the applicant. The applicant needs to verify that the subgrade modulus being used in the analyses represents a rock foundation with shear wave velocity equal to 8000 fps. This is Open Item 3.8.5-3.

Westinghouse Response:

As described in subsection 3.8.5.4.1, the vertical and horizontal stiffness of the springs represents a rock foundation with a shear wave velocity of 8000 feet per second.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.8.5.4-3

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

Summary of Issue:

SRP Section 3.8.5 prescribes the preparation of a design report containing the information listed in Appendix C to SRP 3.8.4. During the April 2 through 5, 2003 audit, the design summary report for the basemat foundation was not available for review by the staff. Completion of the design summary report and review by the staff is Open Item 3.8.5.4-2.

Westinghouse Response:

The basemat design is documented in calculation APP-1010-CCC-001, Rev 1, "AP1000 Basemat Design Report". This document was reviewed during the April 2 through 5, 2003 audit. It describes the overall analyses and the design of the critical sections. The design is also summarized in DCD subsection 3.8.5.4.3. The design of the other sections of the basemat will be included in the as-built summary report prepared by the Combined License applicant as described in DCD subsection 3.8.5.4.2.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.8.5.5-1

Original RAI Number(s): 220.018

Summary of Issue:

The NS overturning moment (moment about the EW or long axis of the basemat) increases by 42.8 percent while the EW overturning moment (moment about the NS or short axis of the basemat) increases by 10.8 percent. The EW base shear increases by only 1.8 percent while the NS base shear increases by 8.1 percent. The reported increases are not consistent with the 9.9 percent increase in the NI mass or 17 percent increase in simple equivalent static overturning moment. In addition, the applicant indicates in its response to RAI 220.018 that the safety factor against overturning for the NS earthquake increases for AP1000, compared to AP600, even though the overturning moment increases by 42.8 percent. Resolution of these apparent discrepancies is Open Item 3.8.5.5-1.

Westinghouse Response:

Table 3.8.5.5-1-1 provides a comparison of the overturning moments and factor of safety against overturning for the AP1000 and AP600. As stated in subsection 3.8.5.5.4 of the AP1000 Design Control Document, "The factor of safety against overturning of the nuclear island during a safe shutdown earthquake ... is evaluated using the static moment balance approach assuming overturning about the edge of the nuclear island at the bottom of the basemat."

Seismic forces are obtained for the AP600 plant from the individual "stick" models associated with the Nuclear Island buildings (coupled Auxiliary and Shield Building; Steel Containment Vessel; and Containment Internal Structures) and four different soil conditions (hard rock, soft rock, soft to medium soil (1 x G_{MAX}, 2 x G_{MAX}). Maximum SSE forces and moments are obtained for the various building stick models at El. 100' and then transferred to the bottom of the basemat (El. 60.5') using the principles of statics. The seismic inertia effects of building masses below 100' are added to the seismic loads for each building. The combined nuclear island seismic forces and moments at El. 60.5' are obtained using the absolute sum method for each of the soil types. Then the maximum SSE load that envelopes all of the design soil profiles for each co-directional response is defined and used in the calculation of the overturning factor of safety. The maximum co-directional responses are considered using the (1.0, 0.4, 0.4) combination method.

Seismic forces are obtained for the AP1000 plant from the seismic time history analysis using the combined Nuclear Island stick model. The seismic reactions relative to the center of containment and column lines (edges of NI basemat) at El. 60.5' are obtained for each time interval. The maximum values are documented and used in the stability calculation. The seismic moments already include the vertical seismic component; therefore, the maximum safe shutdown earthquake vertical force is not considered as a reduction of the resisting moment since it is already included in the seismic induced overturning moment. The missing mass seismic inertia effect, that is predominantly the basemat, is added to the seismic forces and moments.

As stated in the DSER open item the AP1000 moment about the EW axis, relative to the center of containment, ranges from 27% (all soil cases) to 43% (hard rock) larger than the comparable moment for the AP600 plant. The difference is less (10% for all soil cases to 25% for hard rock) when comparing the seismic overturning moment relative to the edges of the Nuclear Island, including the vertical seismic component. The AP1000 NI weight minus the buoyant force is about 15% greater than that for the AP600 plant. The AP1000 overturning resisting moment is also about 15% greater than the AP600 value, since the CG for both plants are about the same. It is therefore expected that the AP1000 seismic overturning factors of safety would be about 5% larger (1.15 / 1.1 = 1.05) than those for the AP600 plant when considering all soil cases. The table confirms this, with the percent difference ranging from 2 to 8%. It is noted that the AP600 overturning factors of safety are based on all soil cases. Also, the inclusion of the vertical seismic component in the seismic moment instead of as a reduction in resisting moment, yields lower factors of safety than those reported in the AP600 DCD (NS Earthquake: 2.0 compared to 1.7; EW Earthquake 1.2 compared to 1.13). From this comparison, the loads and safety factors are consistent with the differences in configuration between the two plants.

Design Control Document (DCD) Revision:

Revise last paragraph of subsection 3.8.5.5.4:

The resisting moment is equal to the nuclear island dead weight, minus ~~maximum safe shutdown earthquake vertical force and~~ buoyant force from ground water table, multiplied by the distance from the edge of the nuclear island to its center of gravity. **The overturning moment is the maximum moment about the same edge from the time history analyses of the nuclear island lumped mass stick model described in subsection 3.7.2.**

PRA Revision:

None

Table 3.8.5.5-1-1
Load Comparisons at El. 60.5' and Factor of Safety against Overturning
Units: kips & ft-kip

Reactions	Hard Rock			Envelope of all Soil Cases	
	AP1000	AP600	Ratio [AP1000/AP 600]	AP600	Ratio [AP1000/AP600]
Moments Relative to Center of Containment					
Vertical	101,495	89,234	1.14	106,443	0.95
Shear NS	99,814	91,368	1.09	100,789	0.99
Moment about NS	8,547,756	7,715,877	1.11	8,633,273	0.99
Shear EW	93,516	91,868	1.02	100,789	0.93
Moment about EW	10,662,959	7,478,927	1.43	8,372,903	1.27
Seismic Overturning Moments Relative to End Boundaries of NI Basemat (1)					
Moment Line 1	14,903,017	12,130,338	1.23	13,921,351	1.07
Moment Line 11	14,791,352	11,965,077	1.24	13,724,218	1.08
Moment Line I	12,639,190	10,361,915	1.22	11,789,606	1.07
Moment West Side of Shield Building	13,644,293	10,816,509	1.26	12,331,869	1.11
Weight minus Buoyant force	205,725	180,390	1.14	–	–
Resisting Moment (2)					
Moment Line 1	26,814,196	23,507,523	1.14	–	–
Moment Line 11	25,851,404	22,672,317	1.14	–	–
Moment Line I	15,419,089	13,372,671	1.15	–	–
Moment West Side of Shield Building	17,702,636	15,670,119	1.13	–	–
Overturning Factors of Safety					
Moment Line 1	1.80	1.94	0.93	1.69	1.07
Moment Line 11	1.75	1.89	0.92	1.65	1.06
Moment Line I	1.22	1.29	0.95	1.13	1.08
Moment West Side of Shield Building	1.30	1.45	0.90	1.27	1.02

(1) Includes vertical seismic component with SSE induced overturning moment
(2) Dead weight and buoyant force are reflected

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 4.5.1-1

Original RAI Number(s): 252.001

Summary of Issue:

The recent experience with VHP nozzle cracking has identified the need for baseline inspection data to determine if an indication is service-induced cracking, or an artifact from fabrication. The staff requested information on what preservice examinations will be performed on the VHP nozzles. In a letter dated April 7, 2003, the applicant responded that preservice examinations for the closure head will include a baseline top-of-the head visual examination, ultrasonic examinations of the inside diameter surface of each vessel head penetration, eddy current examination of the surface of the head penetration welds and the inside diameter surface of the penetrations, and post-hydro liquid penetrant examinations of accessible surfaces that have undergone preservice inspection eddy current examinations. Any indications exceeding the ASME Code Section III requirements would be removed. The information in the RAI response has been provided in DCD Tier 2 Section 5.3.4.7. The information on preservice examinations also needs to be addressed by a COL applicant, and should be reflected in DCD Tier 2 Section 5.3.6, "Combined License Information." This is identified as Open Item 4.5.1-1 and COL Action Item 4.5.1-1.

Westinghouse Response:

The Combined License applicant commitment to specific preservice examinations of the reactor vessel closure head will be added to the existing commitment in DCD section 5.2.6.2 for the Combined License applicant to provide a plant specific preservice inspection program.

Design Control Document (DCD) Revision:

From DCD Revision 5, page 5.2-32:

5.2.6 Combined License Information Items

5.2.6.1 ASME Code and Addenda

The Combined License applicant will address in its application the portions of later ASME Code editions and addenda to be used to construct components that will require NRC staff review and approval. The Combined License applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code edition and addenda added as part of the Combined License application. The Combined License applicant will address the addition of ASME code cases approved subsequent to design certification.

5.2.6.2 Plant Specific Inspection Program

The Combined License applicant will provide a plant-specific preservice inspection and inservice inspection program. The program will address reference to the edition and addenda of the ASME Code Section XI used for selecting components subject to examination, a description of the components exempt from examination by the applicable code, and drawings or other descriptive information used for the examination.

The preservice inspection program will include examinations of the reactor vessel closure head equivalent to those outlined in DCD section 5.3.4.7.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 4.5.1-2

Original RAI Number(s): 252.001

Summary of Issue:

The discovery of leaks and nozzle cracking at the Davis-Besse Nuclear Power Station and other operating PWR plants has made clear the need for more effective inspections of reactor ressure vessel heads and associated penetration nozzles. The current reactor pressure vessel head inspection requirements require visual examination of the insulated surface or surrounding area for signs of leakage. Such inspections have not been sufficient to reliably detect circumferential cracking of reactor pressure vessel head nozzles and corrosion of the reactor pressure vessel head. Circumferential cracking of reactor pressure vessel head nozzles and corrosion of the reactor pressure vessel head pose a safety concern because of the possibility of a nozzle ejection or loss-of-coolant accident if the conditions are not detected and repaired. NRC Order EA-03-009 establishes interim requirements to ensure that current PWR licensees implement and maintain appropriate measures to inspect and, as necessary, repair reactor pressure vessel heads and associated penetration nozzles. This order addresses requirements for both Alloy 600/82/182 materials in the original heads and Alloy 690/52/152 materials in replacement heads and in the AP1000 reactor pressure vessel head design.

Therefore, the staff finds that the COL applicant should perform analyses and inservice inspections and provide reports and notifications equivalent to those contained in Sections IV.A to IV.F of NRC Order EA-03-009, "Interim Inspection Requirements for Reactor Pressure Vessel Heads at PWRs." These activities should include susceptibility calculations and categorization, visual, surface and volumetric examinations, and preparation of reports and notifications. This is identified as Open Item 4.5.1-2 and COL Action Item 4.5.1-2.

Westinghouse Response:

The Combined License applicant commitment to perform inservice inspections of the reactor vessel closure head equivalent to those contained in NRC Order EA-03-009, "Interim Inspection Requirements for Reactor Pressure Vessel Heads at PWRs", will be added to the existing commitment in DCD section 5.2.6.2 for the Combined License applicant to provide a plant specific inservice inspection program.

Design Control Document (DCD) Revision:

From DCD Revision 5, page 5.2-32:

5.2.6 Combined License Information Items

5.2.6.1 ASME Code and Addenda

The Combined License applicant will address in its application the portions of later ASME Code editions and addenda to be used to construct components that will require NRC staff review and approval. The Combined License applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code edition and addenda added as part of the Combined License application. The Combined License applicant will address the addition of ASME code cases approved subsequent to design certification.

5.2.6.2 Plant Specific Inspection Program

The Combined License applicant will provide a plant-specific preservice inspection and inservice inspection program. The program will address reference to the edition and addenda of the ASME Code Section XI used for selecting components subject to examination, a description of the components exempt from examination by the applicable code, and drawings or other descriptive information used for the examination.

The inservice inspection program will address inspections of the reactor vessel closure head contained in NRC Order EA-03-009, "Interim Inspection Requirements for Reactor Vessel Heads at PWRs".

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **5.2.3-1**

Original RAI Number(s): **251.012**

Summary of Issue:

The staff reviewed DCD Tier 2 Section 5.2.3.1, "Materials Specifications," to determine the suitability of the RCPB materials for this application. The AP1000 design conforms with the guidance provided in RG 1.85, "Materials Code Case Acceptability ASME Section III Division 1," and appropriate provisions of the ASME Code.

The staff noted that the DCD states that the RCP pressure housing will be made from SA 351 or SA 352 CF3A material and that the RCP pressure boundary valve bodies may be castings of SA 351 CF3A. In addition, the DCD states that CASS will not exceed a ferrite content of 30 FN. Since CASS RCP pressure boundary components are subject to thermal embrittlement, the staff requested, in RAI 251.012, the applicant discuss the impact of this aging effect on the integrity of these components, how the thermal embrittlement mechanism has been considered in the design and material selection for the RCPB components, and the need to perform inspections to detect this aging effect. In its response, the applicant stated that, based on experience with casting materials, the selection of low carbon grade casting, i.e., CF3A, and control of the material specifications to below 20 FN, there should be no significant impact of thermal aging on the integrity of the components. The applicant responded further that the ASME Code inservice inspections will be relied on to detect the effects of any thermal aging. The proposed DCD change in the response to RAI 251.012 discusses the COL action items regarding these inspections in DCD Tier 2 Section 5.2.6, "Combined License Information Items." The applicant also committed to revising the limit of the ferrite content of CASS to a maximum of 20 FN. This revised FN was provided in Revision 4 of DCD Tier 2 Section 5.2.3.1, "Materials Specifications." The staff reviewed Revision 4 to the DCD and, subject to the clarification discussed below, finds it acceptable since it conforms with the guidance in RG 1.31,"Control of Ferrite Content in Stainless Steel Weld Metal," and criteria acceptable to the staff in the May 19, 2000, letter from C. Grimes to D. Walters.

The applicant needs to clarify in the DCD that the method used to calculate the δ-ferrite is based on Hull's equivalent factors or a method producing an equivalent level of accuracy; i.e., ± 6% deviation between the measured and calculated values, as discussed in the May 19, 2000, letter from C. Grimes to D. Walters. This is Open Item 5.2.3-1.

Westinghouse Response:

DCD section 5.2.3.1 will be revised to document that the calculation of ferrite content is based on Hull's equivalent factors.

Design Control Document (DCD) Revision:

From DCD Revision 5 page 5.2-10, Section 5.2.3-1:

Table 5.2-1 material specifications are the materials used in the AP1000 reactor coolant pressure boundary. The materials used in the reactor coolant pressure boundary conform to the applicable ASME Code rules. Cast austenitic stainless steel does not exceed a ferrite content of 20 FN. Calculation of ferrite content is based on ~~ASTM A800 (Reference 7).~~ **Hull's equivalent factors.**

From DCD Revision 5 page 5.2-32:

5.2.7 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), and WCAP-7907-A (Nonproprietary), April 1984.
2. EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report, Interim Report, April, 1982.
3. Logsdon, W. A., Begley, J. A., and Gottshall, C. L., "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metals," WCAP-9292, March 1978.
4. Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary), March 1970, and WCAP-7735 (Nonproprietary), August 1971.
5. Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1975.
6. Enrietto, J. F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976.

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

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 6.2.1.8.2-1

Original RAI Number(s): 650.004

Summary of Issue:

The applicant's February 21, 2003, response to RAI 650.004 also included an analysis of the IRWST screens' capability to accommodate debris accumulation. The staff's review of the applicant's analysis showed that the mass of resident debris assumed by the applicant (i.e., 227 kg, or 500 lb) was consistent with estimates made for current generation PWRs in the Generic Safety Issue (GSI) 191 parametric study (NUREG/CR-6772). However, the staff could not accept this analysis, primarily because the applicant assumed that a single density value is valid for all density-dependent calculations involving resident fibrous debris. According to the physical properties of analyzed types of fibrous materials, potentially different density values may be required to correctly determine the settling velocity (i.e., the material density), to calculate a volume from the assumed mass (i.e., the "as-found" density), and to determine the thickness and porosity of the associated debris bed (i.e., the rubblized density). As a result of the applicant's single-density assumption, which deviated significantly from the material properties of the low-density fiberglass on which the head loss data referenced by the applicant was based, the NRC staff concluded that the calculation was unacceptable. During a teleconference on April 3, 2003, the applicant agreed to resubmit its response to RAI 650.004. in light of the staff's concern. Pending an acceptable resolution of this concern, the staff considers the capability of the AP1000 IRWST screens to accommodate anticipated debris loadings to be DSER Open Item 6.2.1.8.2-1.

Westinghouse Response:

Westinghouse revised its response to RAI 650.004 in order to address the NRC concerns as discussed in our teleconference on April 3, 2003. The revised RAI response was submitted to the NRC on April 24, 2003 in letter DCP/NRC1580.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **6.2.1.8.3-1**

Original RAI Number(s): **650.001**

Summary of Issue:

The water level in containment following a LOCA would be sufficiently high that DCD Tier 2 Section 3.4.1.2.2.1 states that inventory from the containment pool would "... flow back into the RCS via the break location ...." In light of this statement, the staff issued RAI 650.001 to request additional information concerning the potential for entrained debris to cause blockage at flow restrictions within the RCS once flow begins entering through the break location after flood-up (i.e., bypassing the recirculation screens). In a letter dated February 21, 2003, the applicant responded to RAI 650.001 by submitting an analysis which concluded that RMI debris is incapable of causing such blockage. Although the applicant's response partially addressed the staff's RAI, it was not complete because it did not address the potential for other sources of debris, such as fibrous debris and floatable debris, to enter the RCS through the break location and block requisite core cooling flowpaths. Pending the complete resolution of this concern, the staff considers debris blockage in the RCS to be DSER Open Item 6.2.1.8.3-1.

Westinghouse Response:

Westinghouse revised its response to RAI 650.001 in order to address the NRC concerns as discussed in our teleconference on April 3, 2003. The revised RAI response was submitted to the NRC on April 24, 2003 in letter DCP/NRC1580.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 6.2.1.8.3-2

Original RAI Number(s): 650.006

Summary of Issue:

In RAI 650.006, the staff questioned whether non-safety-related coatings inside the containment could disbond and subsequently block the containment recirculation screens. In a letter dated February 21, 2003, the applicant responded to RAI 650.006 by submitting calculations of the trajectories of settling paint particles to provide confidence that the particles are incapable of passing around the protective screen plate and blocking a significant fraction of the recirculation sump screen surface. The applicant's RAI response further stated that no coating debris can approach the recirculation screens without passing around the protective plates because coatings are not permitted on the surfaces inside the plates. ITAAC commitment 8.c(x) in DCD Tier 1 Table 2.2.3-4 states that the applicant will verify that the dry film density of non-safety-related coating materials is consistent with the assumed value in the settling calculation (i.e., $\geq$ 1600 kg/m^3, or 100 lb/ft^3). The particle sizes and settling rates assumed in the applicant's calculation are similar to or more conservative than those previously accepted by the staff in its review of the AP600 (NUREG-1512) and the Comanche Peak Steam Electric Station Units 1 and 2 (NUREG-0797, Supplement No. 9, dated March 1985). However, according to recent evidence that resident fibrous material may exist in containments and considering operational experience and test data concerning coating failures, the staff considers that paint particles significantly smaller than 200 mils in diameter could become trapped in the interstitial locations of a fibrous debris bed and contribute to the blockage of the recirculation screens. Therefore, in a teleconference on April 3, 2003, the staff requested additional justification from the applicant to support the assumption that paint particles smaller than 200 mils are not a blockage concern for the containment recirculation screens. The staff considers the response to RAI 650.006 to be an open item pending the resolution of this concern. This is DSER Open Item 6.2.1.8.3-2.

Westinghouse Response:

Westinghouse revised its response to RAI 650.006 in order to address the NRC concerns as discussed in our teleconference on April 3, 2003. The revised RAI response was submitted to the NRC on April 24, 2003 in letter DCP/NRC1580.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **6.2.1.8.3-3**

Original RAI Number(s): **650.005**

Summary of Issue:

The staff's review found that insufficient information was available in the DCD to determine whether the containment recirculation screens are capable of tolerating anticipated post-accident debris loadings. Therefore, in RAI 650.005, the staff requested additional information from the applicant to determine the debris-blockage failure criterion of the containment recirculation screens. The applicant responded to RAI 650.005 in a letter dated February 21, 2003, by providing an analysis intended to demonstrate that the AP1000 recirculation screens could accommodate a mass of resident debris (i.e., 227 kg, or 500 lb) that is equivalent to estimates made for current generation PWRs in the GSI-191 parametric study (NUREG/CR-6772). However, the staff could not accept this analysis, primarily because the applicant assumed that a single density value is valid for all density-dependent calculations regarding resident fibrous debris. According to the physical properties of analyzed types of fibrous materials, potentially different density values may be required to correctly determine the settling velocity (i.e., the material density), to calculate a volume from the assumed mass (i.e., the "as-found" density), and to determine the thickness and porosity of the associated debris bed (i.e., the rubblized density). As a result of the applicant's single-density assumption, which deviated significantly from the material properties of the low-density fiberglass on which the head loss data referenced by the applicant was based, the NRC staff concluded that the calculation was unacceptable. During a teleconference on April 3, 2003, the applicant agreed to resubmit its response to RAI 650.005, in light of the staff's concern. Pending an acceptable resolution of this concern, the staff considers the capability of the AP1000 containment recirculation screens to accommodate anticipated debris loadings to be DSER Open Item 6.2.1.8.3-3.

Westinghouse Response:

Westinghouse revised its response to RAI 650.005 in order to address the NRC concerns as discussed in our teleconference on April 3, 2003. The revised RAI response was submitted to the NRC on April 24, 2003 in letter DCP/NRC1580.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **6.2.6.4-1**

Original RAI Number(s): **480.010**

Summary of Issue:

In RAI 480.010, the staff requested that Westinghouse provide justification for why the AP1000 differed from the AP600 in the treatment of P_a (the peak calculated containment internal pressure for the design basis loss of coolant accident) in the Technical Specifications. Westinghouse provided a response to this concern in a letter dated April 11, 2003 (ADAMS Accession No. ML031050025). However, the item remains open, as discussed below.

In a response to RAI 480.010, dated April 11, 2003, Westinghouse provided the following as justification for not placing the numerical value of P_a into the Technical Specifications:

1. It is simpler and reduces future changes to the DCD, and is consistent with the overall TS improvement strategy to minimize the need for a plant license amendment or Bases update for parameters that are expected to change due to re-analysis.

2. It is not clear that Appendix J specifically requires that the numerical value for P_a be included in the Technical Specifications. Appendix J, Option B, states that P_a is specified "...in the Technical Specifications." Westinghouse assumes that this is a reference to the entire Technical Specifications document, which includes the individual technical specifications and the associated bases.

3. The definition of P_a in Option B, "...the calculated peak containment internal pressure related to the design basis loss-of-coolant accident...", is incorrect for AP1000, since the limiting calculated peak containment internal pressure in DCD 6.2 occurs for a steamline break accident.

To resolve the issue, Westinghouse plans to revise the TS Bases to state the numerical value of P_a.

The staff carefully considered the requirements of Appendix J and the objectives of the TS improvement program when developing the latest revision of the Standard Technical Specifications. The staff determined that, despite the inconvenience for future plant-specific license amendments, Appendix J, Option B, requires the numerical value of P_a to be stated in the Technical Specifications, not in the Bases. This is reflected in the Standard Technical Specifications. Westinghouse's proposed resolution of this issue is therefore unacceptable.

Also, Westinghouse's assertion, that the definition of P_a in Option B is incorrect for AP1000, is in error. P_a is not meant to bound the calculated peak containment internal pressures of all postulated accidents. P_a is a parameter specifically established for the purpose of radiological consequence analysis and containment leakage rate testing. For this reason, only accidents that produce a significant radioactive source term in the containment are considered when the value of P_a is determined. Steamline breaks in the AP1000 do not produce a significant radioactive source term in the containment. Of course, containment design pressure must bound the calculated peak containment internal pressures of all postulated accidents, but containment design pressure is not the same as P_a. Thus, the design basis loss-of-coolant accident pressure is the correct parameter for determining the value of P_a. For the reasons stated above, this is DSER Open Item 6.2.6.4-1.

Westinghouse Response:

Technical Specification 5.5.8 will be revised to include the numerical values for the design basis loss-of-coolant accident containment internal pressure.

Technical Specification Bases 3.6.4 will be revised to reflect the definition of P_a as the peak pressure from a LOCA rather than the peak pressure from the limiting DBA. Since the peak pressure for AP1000 is not from a LOCA, the wording of the Standard Technical Specifications must be modified somewhat for this definition.

Design Control Document (DCD) Revision:

Technical Specification 5.5.8 and Technical Specification Bases 3.6.4 will be revised as shown in the attachments.

PRA Revision:

None

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of transients which result in a negative pressure.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the operating band of conditions used in the containment pressure analyses for the Design Basis Events which result in internal or external pressure loads on the containment vessel. Should operation occur outside these limits, the initial containment pressure would be outside the range used for containment pressure analyses.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients (Ref. 1).

The initial pressure ~~containment~~ condition used in the containment analysis was 15.7 psia (1.0 psig). This resulted in a maximum peak pressure from a-LOCA, P_a, of [55.3] psig.~~DBA as indicated in Reference 1.~~ The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure~~, P_a,~~ results from the ~~limiting~~ SLB. The maximum containment pressure resulting from the ~~worst case~~ SLB, ~~of~~ [57.3] psig, does not exceed the containment design pressure, [59] psig.

The containment was also designed for an external pressure load equivalent to 3.0 psig. The limiting negative pressure transient is a loss of all AC power sources coincident with extreme cold weather conditions which cool the external surface of the containment vessel. The initial pressure condition used in this analysis was -0.2 psig. This resulted in a minimum pressure inside containment, as illustrated in Reference 1, which is less than the design load. Other external pressure load events evaluated include:

Failed fan cooler control

Malfunction of containment purge system

5.5 Programs and Manuals

5.5.7 Safety Function Determination Program (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.8 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995," as modified by approved exceptions.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [55.3 psig]. The containment design pressure is 59 psig.~~less than the design pressure of containment. The Applicable Safety Analyses section of the Bases for TS 3.6.4 identifies the peak containment internal pressure for the design basis event and the design containment pressure.~~

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.10% of primary containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for Type A tests;

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq$ [0.05] L_a when tested at $\geq P_a$,

b) For each door, leakage rate is $\leq$ [0.01] L_a when pressurized to $\geq$ [10] psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **8.2.3.1-1**

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

Summary of Issue:

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

Westinghouse Response:

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **9.5.2-1**

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

Summary of Issue:

10CFR 73.55 (e) (f) discusses that placement of backup power supplies for certain communication systems be in vital areas. This is mentioned in DCD Tier 2 Section 13.6, "Security," for "vital equipment," but it is not clear that the "non-portable communication equipment" specified in 10CFR 73.55 (f) is vital equipment. The DCD should clarify the categorization of communication equipment and the requirement addressing this equipment in 10CFR 73.55 (f). This is Open item 9.5.2-1.

Westinghouse Response:

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **9.5.2-2**

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

Summary of Issue:

10 CFR 73.55 (g) mentions testing requirements for certain communication systems. This has not been addressed in the DCD. Therefore, this is Open Item 9.5.2-2.

Westinghouse Response:

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 9.5.2-4

Original RAI Number(s): 420.048c

Summary of Issue:

SRP Section 9.5.2 provides reviewer guidance on the design of communication systems (i.e intra-pant and plant to offsite). Part of that guidance states, "Communications system will be protected from EMI/RFI effects of other plant equipment and there will be adequate testing and field measurements where necessary to demonstrate effective communications." In addition, SRP Section 9.5.2 discusses the general requirement that addresses the need for communication equipment to provide effective communication during the "full spectrum of ...conditions ...under maximum potential noise levels."

The staff believes the DCD has not sufficiently covered communication testing for plant startup and operations in sufficient detail to facilitate understanding of how effective communications will be demonstrated including EMI/RFI effects on the equipment. The staff also believe the DCD has not sufficiently addressed how effective communications will be sustained for maximum potential noise levels. This is Open Item 9.5.2-4.

Westinghouse Response:

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

Design Control Document (DCD) Revision:

See response to RAI 420.048 (item c) transmitted by Westinghouse letter DCP/NRC1590, dated May 14, 2003.

PRA Revision:

None

DSER Open Item Number: 14.2.7-1

Original RAI Number(s): 261.014

Summary of Issue:

In DCD Appendix 1A, the applicant provided the following information related to RG 1.41:

> The guidelines are followed for Class 1E dc [direct current] power systems during the preoperational testing of the AP1000 redundant onsite electric power systems to verify proper load group assignments, except as follows. Complete preoperational testing of the startup, sequence loading, and functional performance of the load groups is performed where practical. In those cases where it is not practical to perform complete functional testing, an evaluation is used to supplement the testing.

The NRC staff lacked sufficient information to determine if this exception to RG 1.41 was acceptable. Specifically, the staff was unable to identify which regulatory position in RG 1.41 the exception applied to, if the exception applied to both alternating current (ac) and dc systems, and in what cases was it not practical to perform functional testing. Therefore, in Request for Additional Information (RAI) 261.014, the NRC staff requested

Westinghouse Response:

A response to this issue was provided in response to RAI 261.014 transmitted by Westinghouse letter DCP/NRC1590, dated May 14, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

DSER Open Item Number: 14.2.7-3

Original RAI Number(s): None

Summary of Issue:

The NRC staff believes that the applicant should delete this exception to RG 1.68 in WCAP-15799 and state that the test abstract in DCD Tier 2 Section 14.2.10.4.28 meets the guidance in RG 1.68, Regulatory Position C.1, Appendix A.5, Test 5.d.d, or provide additional information to clarify this exception. This is DSER Open Item 14.7.2-3.

Westinghouse Response:

Westinghouse agrees that the test abstract in DCD Tier 2 meets the guidance in RG 1.68, Regulatory Position C.1, Appendix A.5, test d.d. WCAP-15799 will be revised as shown below.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

WCAP-15799 Section 14 will be revised as shown:

INITIAL TEST PROGRAM

Criteria Section	Reference Criteria	AP1000 Position	Comments/Summary of Exceptions
SRP 14.2 - Initial Plant Test Program (Rev. 2, 7/81)			
1.	R.G. 1.68 C.3	N/A	Not applicable to AP1000 Design Certification. This is the Combined License applicant's responsibility.
2.	R.G. 1.68 C.4	N/A	Not applicable to AP1000 Design Certification. This is the Combined License applicant's responsibility.
3.	R.G. 1.68 C.1, App. A.1.a	Acceptable	Applies to AP1000 RCS components. (Jet pumps are applicable to BWRs only.)
	R.G. 1.68 C.1, App. A.1.b	Acceptable	Applies to the AP1000 reactivity control system, except the systems for BWRs such as rod worth minimizers.
	R.G. 1.68 C.1, App. A.1.c	Acceptable	
	R.G. 1.68 C.1, App. A.1.d	Exception	These systems have been eliminated due to the design of the AP1000 passive safety systems. The functions of these systems are replaced by the PRHR heat exchangers of the passive core cooling system.
	R.G. 1.68 C.1, App. A.1.e	Acceptable	
	R.G. 1.68 C.1, App. A.1.f	Acceptable	
	R.G. 1.68 C.1, App. A.1.g(1)	Acceptable	

Criteria Section	Reference Criteria	AP1000 Position	Comments/Summary of Exceptions
	R.G. 1.68 C.1, App. A.1.g(2)	Acceptable	
	R.G. 1.68 C.1, App. A.1.g(3)	Exception	The AP1000 nonsafety-related diesel-generators are not required for safe shutdown.
	R.G. 1.68 C.1, App. A.1.g(4)	Acceptable	
	R.G. 1.68 C.1, App. A.1.h	Acceptable	The characteristics of the AP1000 passive safety systems allow the support systems such as the cooling water systems, the HVAC and the ac power sources to be nonsafety-related and simplified. The capability of these systems is established by testing.
	R.G. 1.68 C.1, App. A.1.i	Acceptable	The AP1000 has no secondary containment. Therefore, this guideline applies only to primary containment.
	R.G. 1.68 C.1, App. A.1.j-o	Acceptable	
	R.G. 1.68 C.1, App. A.2	Acceptable	As applicable for PWR.
	R.G. 1.68 C.1, App. A.3	Acceptable	As applicable for PWR.
	R.G. 1.68 C.1, App. A.4	Acceptable	As applicable for PWR.
	R.G. 1.68 C.1, App. A.5	**Acceptable**~~Exception~~	**As applicable for PWR.**~~Since the remote shutdown workstation is similar to the main control room work stations, it is unnecessary to perform a pre-operational test to place the plant in a safe shutdown condition and maintain it there from the remote shutdown workstation. Remote shutdown capability testing is performed by testing of the controls and indications of the remote shutdown workstation and separately demonstrating the ability of the PRHR system to maintain safe shutdown.~~

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **14.3.2-1**

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

Summary of Issue:

Section 2.2.1, "Containment System." The staff finds that item No. 2 under the Design Description for the containment system states that the components identified in Table 2.2.1-1 and the piping identified in Table 2.2.1-2 are designed and constructed in accordance with ASME Code Section III requirements. However, during the April 2-5, 2003, design audit, the staff found that the applicant did not complete the final analyses and design of the containment vessel, including attached components and piping systems (Section 3.8.2.1 of this report). The issue related to the containment design is designated as Open Item 14.3.2-1.

Westinghouse Response:

See the Westinghouse response to DSER Open Item 3.8.2.1-1. Westinghouse identified additional detailed analyses to be performed for the containment vessel in letter DCP/NRC1583, dated May 1, 2003. These analyses are available for NRC staff review and demonstrate that the AP1000 containment vessel satisfies the acceptance criteria documented in the DCD.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-2

Original RAI Number(s): None

Summary of Issue:

Section 2.2.1. The phrase "structural integrity and" should be added in two places: (1) Design Description No. 5 for the containment system, and (2) Sub-item No. 5.ii under the Acceptance Criteria of ITAAC Table 2.2.1-3. The sentence should read "...the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of structural integrity and safety function." This is Open Item 14.3.2-2.

Westinghouse Response:

Westinghouse will revise the ITAAC as requested.

Design Control Document (DCD) Revision:

Tier 1 section 2.2.1 will be revised as follows:

5. The seismic Category I equipment identified in Table 2.2.1-1 can withstand seismic design basis loads without loss of **structural integrity and** safety function.

Tier 1 Table 2.2.1-3 will be revised as follows:

5. The seismic Category I equipment identified in Table 2.2.1-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.2.1-1 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.2.1-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of **structural integrity and** safety function. iii) The as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-3

Original RAI Number(s): None

Summary of Issue:

Section 2.2.1. The thickness of the steel containment vessel should be designated as Tier 1 information and specified in Section 2.2.1 or listed in Table 3.3-1. This is Open Item 14.3.2-3.

Westinghouse Response:

The thickness of the steel containment vessel is not identified as Tier 1 information. This is consistent with the other certified designs including the AP600. The acceptability of the steel containment vessel is demonstrated by the ASME Design Report in ITAAC 2.a). The thickness is specified in subsection 3.8.2 as Tier 2 information. This information will be changed to Tier 2* for the AP1000.

Design Control Document (DCD) Revision:

Figure 3.8.2-1 shows the containment vessel outline, including the plate configuration and crane girder. It is a free-standing, cylindrical steel vessel with ellipsoidal upper and lower heads. *[The containment vessel has the following design characteristics:*

Diameter: 130 feet
Height: 215 feet 4 inches
Design Code: ASME III, Div. 1
Material: SA738, Grade B
Design Pressure: 59 psig
Design Temperature: 300°F
Design External Pressure: 2.9 psid

*The wall thickness in most of the cylinder is 1.75 inches.]** The wall thickness of the lowest course of the cylindrical shell is increased to 1.875 inches to provide margin in the event of corrosion in the embedment transition region. *[The thickness of the heads is 1.625 inches.]** The heads are ellipsoidal with a major diameter of 130 feet and a height of 37 feet, 7.5 inches.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-4

Original RAI Number(s): None

Summary of Issue:

Section 2.3.2, "Chemical and Volume Control System." The staff found that incomplete design commitments related to controls and displays exist in the current system-based ITAAC. For example, one current description states that, "[c]ontrols exist in the MCR [main control room] to cause the pumps identified in Table 2.3.2-3, to perform the listed function." The staff recommends revising this design commitment to indicate that not only should the controls exist in the MCR and perform their intended functions, but the controls should be designed so that they are usable by operators. A suggested revision to accommodate this change is, "[c]ontrols exist in the MCR to cause the pumps identified in Table 2.3.2-3 to perform the listed function and are designed in accordance with state-of-the-art human factors principles as required by 10 CFR 50.34(f)(2)(iii)." The same concern applies to the current design commitment statements related to displays. As an example, the current design commitment of "[s]afety-related displays identified in Table 2.3.2-1 can be retrieved from the MCR," should be changed to "[s]afety-related displays identified in Table 2.3.2-1 can be retrieved from the MCR, perform their intended function, and are designed in accordance with state-of-the-art human factors principles as required by 10 CFR 50.34(f)(2)(iii)." These recommended changes to the above-cited examples apply to other current design commitments for system-based ITAAC. This is Open Item 14.3.2-4.

Westinghouse Response:

Westinghouse does not agree that the proposed words need to be added to the system-based Tier 1 sections. DCD Tier 1 Section 3.2 is the appropriate place to have the design commitments that demonstrate compliance with 10 CFR 50.34(f)(2)(iii). DCD Section 3.2 stipulates that MCR controls are designed in accordance with state-of-the-art human factor principles. This applies to each of the individual controls covered by the various system ITAAC; therefore, this commitment satisfies the requirement.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-5

Original RAI Number(s): None

Summary of Issue:

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

Westinghouse Response:

Based on a review of the AP600 heavy load analyses for the equipment and maintenance hatch hoists, Westinghouse has revised the classification of the AP1000 maintenance hatch hoist to a non-single failure proof design which is consistent with the AP600 classification. Coincident with this change Westinghouse has revised the associated ITAAC to delete the design commitment for the maintenance hatch hoist and to provide a design commitment for the equipment hatch hoist related to the single failure nature of the design. These changes make the DCD Tier 1 and Tier 2 information on the maintenance and equipment hatch hoists consistent.

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

Design Control Document (DCD) Revision:

The following changes have been incorporated into DCD Revision 5.

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

2.3.5 Mechanical Handling System

Design Description

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

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

1. The functional arrangement of the MHS is as described in the Design Description of this Section 2.3.5.

2. The seismic Category I equipment identified in Table 2.3.5-1 can withstand seismic design basis loads without loss of safety function.

3. The MHS provides the following safety-related functions:

   a) The containment polar crane prevents the uncontrolled lowering of a heavy load.

   b) The equipment hatch hoist **prevents the uncontrolled lowering of a heavy load.**~~, positions the hatch to minimize loss of water inventory from containment during loss of shutdown cooling events.~~

   ~~c) The maintenance hatch hoist positions the hatch to minimize loss of water inventory from containment during loss of shutdown cooling events.~~

4. The spent fuel shipping cask crane cannot move over the spent fuel pool.

Inspections, Tests, Analyses, and Acceptance Criteria

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

Table 2.3.5-1

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

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

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

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

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

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

Table 3.2-3 (Sheet 8 of 67)

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

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

From DCD page 9.1-37:

9.1.5.1 Design Basis

9.1.5.1.1 Safety Design Basis

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

From DCD page 9.1-38:

9.1.5.2 System Description

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

From DCD page 9.1-41:

9.1.5.3 Safety Evaluation

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

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

The polar crane and the equipment ~~and maintenance~~ hatch hoist systems are single failure proof. These systems stop and hold a critical load following the credible failure of a single component. Redundancy is provided for load bearing components such as the hoisting ropes, sheaves, equalizer assembly, hooks, and holding brakes. These systems are designed to support a critical load during and after a safe shutdown earthquake. The equipment and maintenance hatch hoist systems are designed to remain operational following the event. The polar crane is designed to withstand rapid pressurization of the containment during a design basis loss of coolant accident or main steam line break, without collapsing.

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

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

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

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-7

Original RAI Number(s): 420.048d

Summary of Issue:

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

Westinghouse Response:

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-9

Original RAI Number(s): 252.010

Summary of Issue:

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

Westinghouse Response:

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **14.3.2-10**

Original RAI Number(s): **252.011**

Summary of Issue:

Operating experience continues to show cracking of Alloy 600 components. Recent experience appears to indicate that cracking has even occurred in welds or components not previously expected to crack, based on the temperature of the weld or component and the time in service. The staff believes that the use of Alloy 690 materials in contact with reactor coolant is a substantial improvement over the use of materials currently in wide use in the industry. However, data is not currently available to demonstrate that cracking in these welds and components will not occur over the projected 60-year design lifetime of an AP1000 plant. The staff also believes that bare metal visual inspection of these locations is highly effective in identifying locations where cracking occurs. Please provide information to describe the extent to which the insulation of all Alloy 600/690 components and welds in the reactor coolant pressure boundary (not just upper reactor vessel head penetrations) will be designed to readily facilitate bare metal visual inspection during refueling outage conditions. Please provide proposed ITAAC to verify that all Alloy 600/690 components and welds in the reactor coolant pressure boundary are identified and are readily accessible for bare metal visual inspection. This is Open Item 14.3.2-10.

Westinghouse Response:

Westinghouse provided the response to this Open Item in the response to RAI 252.011 which was transmitted to the NRC via Westinghouse letter DCP/NRC1592 (dated May 21, 2003).

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-11

Original RAI Number(s): None

Summary of Issue:

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

Westinghouse Response:

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

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

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **14.3.2-13**

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

Summary of Issue:

Section 3.3, "Buildings." Item 2.a of the Design Description and Table 3.3-6 states that the nuclear island structures, including the critical section listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design-basis loads (including seismic loads), as specified in the design description, without loss of structural integrity and the safety related functions. However, as identified in Open Items 3.7.2.3-1, 3.7.2.3-3 and 3.8.5.4-1, the applicant did not demonstrate that the foundation mat will not lift up, and/or the shear walls will not crack during a postulated seismic event. The phenomenon of the foundation mat uplifting and shear wall cracking will directly affect the design adequacy of the nuclear island structures, systems and components, including the thickness of structural elements listed in Table 3.3-1 and safety-related piping systems. Consequently, this is open item 14.3.2-13.

Westinghouse Response:

See the Westinghouse response to DSER Open Item 3.7.2.3-1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-1

Original RAI Number(s): None

Summary of Issue:

Section 2.5.1, "Diverse Actuation System," Table 2.5.1-1, "Functions Automatically Actuated by the DAS" should be modified to include "actuate core makeup tanks, and trip the reactor coolant pumps on low wide-range steam generator water level." This comment is based on the review of The Tier 2 Section 7.7.1.11, "Diverse Actuation System." This is Open Item 14.3.3-1.

Westinghouse Response:

Westinghouse agrees to revise DCD Tier 1 Table 2.5.1-1 as shown below.

Design Control Document (DCD) Revision:

Table 2.5.1-1 Functions Automatically Actuated by the DAS
1. Reactor and Turbine Trip on Low Wide-range Steam Generator Water Level or Low Pressurizer Water Level
2. Passive Residual Heat Removal (PRHR) Actuation and In-containment Refueling Water Storage Tank (IRWST) Gutter Isolation on Low Wide-range Steam Generator Water Level or on High Hot Leg Temperature
3. Core Makeup Tank (CMT) Actuation and Trip All Reactor Coolant Pumps on Low Wide-Range Steam Generator Water Level or Low Pressurizer Water Level
4. Isolation of Selected Containment Penetrations and Initiation of Passive Containment Cooling System (PCS) on High Containment Temperature

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-2

Original RAI Number(s): None

Summary of Issue:

Section 2.5.1 design description item 2(c) should be modified to include "the DAS manual control bypasses the protection and safety monitoring system cabinets." This comment is based on the review of the Tier 2 Section 7.7.1.11, "Diverse Actuation System." This is Open Item 14.3.3-2.

Westinghouse Response:

Westinghouse agrees to revise DCD Tier 1 Section 2.5.1 as shown below.

Design Control Document (DCD) Revision:

2.5.1 Diverse Actuation System

Design Description

The diverse actuation system (DAS) initiates reactor trip, actuates selected functions, and provides plant information to the operator.

The component locations of the DAS are as shown in Table 2.5.1-5.

1. The functional arrangement of the DAS is as described in the Design Description of this Section 2.5.1.

2. The DAS provides the following nonsafety-related functions:

   a) The DAS provides an automatic reactor trip on low wide-range steam generator water level or on low pressurizer water level separate from the PMS.

   b) The DAS provides automatic actuation of selected functions, as identified in Table 2.5.1-1, separate from the PMS.

   c) The DAS provides manual initiation of reactor trip and selected functions, as identified in Table 2.5.1-2, separate from the PMS. These manual initiation functions are implemented in a manner that bypass the control room multiplexers, **the protection and safety monitoring system cabinets**, and the signal processing equipment of the DAS.

   d) The DAS provides main control room (MCR) displays of selected plant parameters, as identified in Table 2.5.1-3, separate from the PMS.

Table 2.5.1-4 Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
2.c) The DAS provides manual initiation of reactor trip, and selected functions, as identified in Table 2.5.1-2, separate from the PMS. These manual initiation functions are implemented in a manner that bypass the control room multiplexers, **the protection and safety monitoring system cabinets**, and the signal processing equipment of the DAS.	Electrical power to the control room multiplexers and PMS equipment will be disconnected and the outputs from the DAS signal processing equipment will be disabled. While in this configuration, an operational test of the as-built system will be performed using the DAS manual actuation controls.	i) The field breakers of the control rod motor-generator sets open after reactor and turbine trip manual initiation controls are actuated. ii) DAS output signals are generated for the selected functions, as identified in Table 2.5.1-2, after manual initiation controls are actuated.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-3

Original RAI Number(s): None

Summary of Issue:

Section 2.5.1 design description item 3(e) should be modified to include "The DAS uses sensors that are separate from those being used by the PMS and the plant control system." This comment is based on the review of the Tier 2 Section 7.7.1.11, "Diverse Actuation System." This is Open Item 14.3.3-3.

Westinghouse Response:

Westinghouse agrees to revise DCD Tier 1 Section 2.5.1 as shown below.

Design Control Document (DCD) Revision:

2.5.1 Diverse Actuation System

Design Description

3. The DAS has the following features:

   a) The signal processing hardware of the DAS uses input modules, output modules, and microprocessor boards that are different than those used in the PMS.

   b) The display hardware of the DAS uses a different display device than that used in the PMS.

   c) Software used in the DAS uses an operating system and a programming language that are different than those used in the PMS.

   d) The DAS has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency (RFI), and electrostatic discharge (ESD) conditions that exist where the DAS equipment is located in the plant.

   e) The sensors identified on Table 2.5.1-3 are used for DAS input **and are separate from those being used by the PMS and plant control system**

Table 2.5.1-4 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
3.e) The sensors identified on Table 2.5.1-3 are used for DAS **input and are separate from those being used by the PMS and plant control system.**	Inspection of the as-built system will be performed.	The sensors identified on Table 2.5.1-3 are used **by DAS and are separate from those being used by the PMS and plant control system.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-4

Original RAI Number(s): None

Summary of Issue:

Section 2.5.2, "Protection and Safety Monitoring System," Table 2.5.2-1 and Figure 2.5.2-1 should be modified to include "two divisions of safety-related postaccident parameter displays" to be consistent with the Tier 1 Section 2.5.2 design description. This is Open Item 14.3.3-4.

Westinghouse Response:

Westinghouse agrees to revise DCD Tier 1 Table 2.5.1-1 as shown below. Figure 2.5.2-1 does not need to be revised. The displays addressed by this open item are included in Figure 2.5.2-1 in the box labeled "Safety-Related Displays and Indications."

Design Control Document (DCD) Revision:

Table 2.5.2-1
PMS Equipment Name and Classification

Equipment Name	Seismic Cat. I	Class 1E	Qual. for Harsh Envir.
PMS Cabinets, Division A	Yes	Yes	No
PMS Cabinets, Division B	Yes	Yes	No
PMS Cabinets, Division C	Yes	Yes	No
PMS Cabinets, Division D	Yes	Yes	No
Reactor Trip Switchgear, Division A	Yes	Yes	No
Reactor Trip Switchgear, Division B	Yes	Yes	No
Reactor Trip Switchgear, Division C	Yes	Yes	No
Reactor Trip Switchgear, Division D	Yes	Yes	No
MCR/RSW Transfer Panels	Yes	Yes	No
MCR Safety-related Displays, Division B	Yes	Yes	No
MCR Safety-related Display, Division C	Yes	Yes	No
MCR Safety-related Controls	Yes	Yes	No

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **14.3.3-5**

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

Summary of Issue:

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

Westinghouse Response:

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

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

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

Design Control Document (DCD) Revision:

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

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-6

Original RAI Number(s): None

Summary of Issue:

Section 2.5.2 design description Item 8(b) should be modified to clarify that the control transfer function is implemented by multiple transfer switches. Each individual transfer switch is associated with only a single safety-related or single non-safety-related group. The ITAAC table should reflect this feature. This is Open Item 14.3.3-6.

Westinghouse Response:

Westinghouse agrees to revise DCD Tier 1 Section 2.5.1 as shown below.

Design Control Document (DCD) Revision:

2.5.1 Diverse Actuation System

Design Description

8. The PMS, in conjunction with the operator workstations, provides the following functions:

   a) The PMS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.2-5. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved in the main control room (MCR). The fixed position controls listed with a "Yes" in the "Control" column are provided in the MCR.

   b) The PMS provides for the transfer of control capability from the MCR to the remote shutdown workstation (RSW) **using multiple transfer switches. Each individual transfer switch is associated with only a single safety-related group or with nonsafety-related control capability.**

   c) Displays of the open/closed status of the reactor trip breakers can be retrieved in the MCR.

Table 2.5.1-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.b) The PMS provides for the transfer of control capability from the MCR to the RSW **using multiple transfer switches. Each individual transfer switch is associated with only a single safety-related or single nonsafety-related group.**	**i) An inspection will be performed to verify that a transfer switch exists for each safety-related division and the nonsafety-related control capability.** ii) An operational test of the as-built system will be performed to demonstrate the transfer of control capability from the MCR to the RSW.	**i) A transfer switch exists for each safety-related division and the nonsafety-related control capability.** ii) Actuation of **each** ~~the~~ transfer switches results in an alarm in the MCR and RSW, the activation of operator control capability from the RSW, and the deactivation of operator control capability from the MCR **for the associated safety-related division or nonsafety-related control capability.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-7

Original RAI Number(s): None

Summary of Issue:

Section 2.5.2, Table 2.5.2-7, "PMS Interlocks," should be modified to include "Interlocks for the Accumulator Isolation Valves and IRWST Discharge Valve" to be consistent with Tier 2 information provided in Section 7.6.2.3. This is Open Item 14.3.3-7.

Westinghouse Response:

The interlocks for the accumulator isolation valve and IRWST discharge valve are not PMS functions and therefore are not included in the PMS ITAAC (DCD Tier 1 Table 2.5.2-7). DCD Tier 2 subsection 7.6.2.3 describes this interlock. As stated in the last sentence of this subsection the confirmatory open and automatic open signals are provided by the plant control system.

This function should not be added to the Tier 1 requirements for the plant control system. The function of assuring that these valves are open whenever these injection paths are required is provided by the Technical Specifications LCOs 3.5.1, 3.5.6, and 3.5.7 (see DCD Tier 2 subsection 7.6.2.3 first bullet).

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-8

Original RAI Number(s): None

Summary of Issue:

Tier 1 Section 2.5.2, Table 2.5.2-6, "PMS Blocks," should be modified to include (1) block automatic rod withdrawal (P-17) and (2) block automatic safeguards (P-4). This comment is based on the review of the Tier 2 Table 7.2-3, "Reactor Trip Permissives and Interlocks," and Table 7.3-2, "Interlocks for Engineered Safety Features Actuation System." This is open item 14.3.3-8.

Westinghouse Response:

Westinghouse agrees to revise DCD Tier 1 Table 2.5.2-6 as shown below.

Design Control Document (DCD) Revision:

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

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-9

Original RAI Number(s): None

Summary of Issue:

Section 2.5.2, Table 2.5.2-8, ITAAC 7(c) columns do not have sufficient criteria to verify that the design commitment is met. Removal of power of non-safety components and review of gateway filtering is not enough. The language should be consistent with acceptance criteria for other ITAACs in this section such as 7(a) and 7(b). A report should be prepared to cover major design considerations such as quality of components, performance requirements, reliability, control access, single-failure criterion, independence, failure modes, testing, and electromagnetic interference/radio frequency interference (EMI/RFI) susceptibility. SRP 7.9 (data communications) may be used as guidance. This is Open Item 14.3.3-9.

Westinghouse Response:

Westinghouse agrees to revise DCD Tier 1 Table 2.5.2-8, item 7c, as shown below.

Design Control Document (DCD) Revision:

Table 2.5.2-8 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
7.c) Data communication between safety and nonsafety systems does not inhibit the performance of the safety function.	**Type tests, analyses, or a combination of type tests and analyses of the PMS gateways will be performed.**~~i) Inspection of the as-built PMS gateways will be performed.~~ ~~ii) An operational test of the as-built PMS gateways will be performed. Power will be removed from the nonsafety components that communicate with the gateways. Real or simulated signals will be used. The automatic and manual actions listed in Tables 2.5.2-2, 2.5.2-3, and 2.5.2-4 will be tested.~~ ~~iii) An operational test of the as-built PMS gateways will be performed. Attempts will be made to send messages to the PMS from the DDS. Some of the messages will be from a predefined list of valid commands and some will not be on the list.~~	**A report exists and concludes that data communication between safety and nonsafety systems does not inhibit the performance of the safety function.**~~i) Each network interface between safety and nonsafety systems includes a buffering circuit.~~ ~~ii) With power removed from the nonsafety components, appropriate PMS output signals are generated after the test or manual actuation signal reaches the specified limit.~~ ~~iii) The gateways filter the incoming message streams and accept only commands from a predefined list of valid commands. All other messages are discarded.~~

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.3-10

Original RAI Number(s): None

Summary of Issue:

Section 2.5.2, Table 2.5.2-8, ITAAC 7(d) columns may not be sufficient to verify the design commitment, especially the terminology "non-class 1E controls" in the performance of the operational tests. The language should be similar to other ITAACs in this section such as 7(a) and 7(b). A report should be prepared to cover the verification process to ensure that no potential signal from the non-safety system that will prevent the PMS from performing it's safety function. This is Open Item 14.3.3-10.

Westinghouse Response:

Westinghouse agrees to revise DCD Tier 1 Table 2.5.2-8, item 7d, as shown below.

Design Control Document (DCD) Revision:

Table 2.5.2-8 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
7.d) The PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.	**Type tests, analyses, or a combination of type tests and analyses of the PMS manual control circuits and algorithms will be performed.**~~i) An operational test of the as-built PMS will be performed. Real or simulated signals will be used. An attempt will be made to block an automatic action as listed in Tables 2.5.2-2 and 2.5.2-3 using the non-Class 1E controls.~~ ~~ii) An operational test of the as-built PMS will be performed using the PMS manual actuation controls. An attempt will be made to block a manual action as listed in Table 2.5.2-4 using the non-Class 1E controls.~~	**A report exists and concludes that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.**~~i) Appropriate PMS output signals are generated after the test signal reaches the specified limit. These output signals remain following an attempt to block these signals using the non-Class 1E controls.~~ ~~ii) PMS output signals are generated for manually actuated functions as identified in Table 2.5.2-4 after the manual actuation controls are operated. These output signals remain following an attempt to block these signals using the non-Class 1E controls.~~

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 16.2-1

Original RAI Number(s): None

Summary of Issue:

The AP1000 Technical Specification (TS) 3.4.10 on reactor coolant system specific activity omits standard technical specification (STS) surveillance requirement (SR) 3.4.16.3. This surveillance requires, on a 184-day frequency, determining $\bar{E}$, the average disintegration energy, from a sample taken in Mode 1 after a minimum of 2 effective full power days and 20 days of Mode 1 operation have elapsed since the reactor was last subcritical for ≥48 hours. Although $\bar{E}$ is not used in the AP1000 TS, Westinghouse should explain why an equivalent surveillance using dose equivalent Iodine -131 is not proposed. This is identified as draft safety evaluation report (DSER) Open Item 16.2-1

Westinghouse Response:

Surveillance requirement 3.4.16.3 will be incorporated into the AP1000 Technical Specifications. However, consistent with the AP1000 approach elsewhere, this surveillance will be based upon noble gas concentration (Xe-133) rather than I-131.

Design Control Document (DCD) Revision:

Incorporate new surveillance requirement 3.4.10.3, as shown in the attached mark-up.

PRA Revision:

None.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.10.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 280 μCi/gm.	7 days
SR 3.4.10.2	- NOTE - Only required to be performed in MODE 1. Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 to 6 hours after a THERMAL POWER change of ≥ 15% of RTP within a 1 hour period
SR 3.4.10.3	- NOTE - Note required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. Determine DOSE EQUIVALENT XE-133 from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.1.10.1-1

Original RAI Number(s): 720.035

Summary of Issue:

In Chapter 26 of the API000 PRA, a design option for the PMS in addition to the one modeled in the PRA is proposed. The option to use the Common Qualified Platform (Common Q) is proposed because of the rapid changes that are taking place In the digital computer and graphic display technologies employed in the modern human systems interface. The applicant assumes that the use of the Common Q option, in the place of the PRA PMS model, does not have any impact on the design certification process because such a process focuses upon the process used to design and implement instrumentation and control systems for the AP1000 rather than on the specific implementation. The staff requested the applicant (see RAI 720.035) to explain the process that will be used to verify that a PMS designed with the "Common Q" option will have equivalent or better reliability than the system modeled in the PRA and how the introduction of the "Common Q" option will affect important PRA-based insights about the PMS.

In its response to RAI 720,035, the applicant asserted that the PRA results are not sensitive to small changes in PMS failure probabilities and that the general architecture of the Common Q PMS is similar to that modeled in the AP1000 PRA. In addition, the applicant stated that the AP600 I&C functional requirements, which have received design certification, will be retained to the maximum extent compatible with the Common Q hardware and software. Also, it is stated that although the details of the AP1000 PRA model follow the AP600 design, the Common Q hardware and software provide a degree of redundancy that is equivalent to the redundancy modeled in the AP1000 PRA. However, the staff believes that further clarification of this issue is needed to ensure that the "Common Q" option will have the same or better reliability than the PMS design modeled in the PRA. A comparison of important features between the "Common Q" option and the PMS modeled in the PRA could help clarify this issue. This comparison should include features found by the PRA to be important contributors to the assumed high reliability of the PMS. Such a comparison, may identify the need for additional or different "design certification requirements" for the "Common Q" option" of PMS. This is Open Item 19.1.10.1-I.

Westinghouse Response:

Additional information on this issue was provided in response to RAI 720.035, Revision 1, transmitted by Westinghouse letter DCP/NRC1557, dated March 26, 2003.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.1.10.1-2

Original RAI Number(s): 720.038

Summary of Issue:

PRA Input to Design Certification Process:

An important objective of the AP1000 design certification PRA is to identify important PRA insights and assumptions and make sure that they are addressed in the design certification through "design certification requirements," such as requirements for ITAAC, the requirement for a D-RAP and COL action items. These requirements will be incorporated in the DCD to ensure that any future plant which references the design will be built and operated in a manner that is consistent with important assumptions made in the design certification PRA.

In its response to RAI 720.038, the applicant provided a preliminary and, recently, a revised list of "design certification requirements." The staff expects the final list of "design certification requirements" to be in agreement with the resolution of all open items identified in the AP1000 DSER The staff is still reviewing the list of "design certification requirements" proposed by the applicant, especially in light of assumptions and insights related to differences in PRA models between the AP600 and API000 designs (e.g., differences in assumptions made in the fire risk analysis), The staff expects the applicant to continue providing requested information to ensure that all important assumptions made in the design certification PRA are appropriately included in the final list of design certification requirements. This is Open Item 19.1.10.1-2

Westinghouse Response:

The PRA insights and assumptions are addressed in the Section 19.59.10 PRA Input to Design Certification Process of the DCD Chapter 19 revision 3 and in the Section 59.10 PRA Input to Design Certification Process of the PRA Chapter 59 revision 1.

Westinghouse believes that the important assumptions made in the design certification PRA are included in the final list of design certification requirements.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.4-1

Original RAI Number(s): 720.060

Summary of Issue:

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

Westinghouse Response:

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.3-1

Original RAI Number(s): None

Summary of Issue:

Major SMA Model Assumptions

The applicant has used a PRA based seismic margin analysis method similar to the AP600 plant. In conducting its SMA, the applicant made the following assumptions:

- Seismic events occur at full power
- The review level earthquake (RLE) is 05 g
- The loss of offsite power occurs at the RLE. No credit is taken for non-safety related diesel generators for on-site AC power
- No credit is taken for non-safety related systems
- Initiating seismic event categories are derived from the AP600 model and the min-max method was used to calculate the plant HCLPF value

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

Westinghouse Response:

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

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

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.3-3

Original RAI Number(s): None

Summary of Issue:

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

Westinghouse Response:

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

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

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

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

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

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 20.7-2

Original RAI Number(s): None

Summary of Issue:

Generic Letter 85-06 provides the explicit QA guidance required by 10 CFR 50.62 for the non-safety related SSCs required to mitigate an ATWS event per 10 CFR 50.62(c)(1). The NRC staff reviewed DCD Tier 2 Sections 15.8, 17.3, Table 17-1, and WCAP-15985, Revision 1, for applicant's resolution of this generic issue.

In DCD Tier 2 Section 15.8, the applicant stated that the AP1000 diverse actuation system (DAS) provides the ATWS mitigation systems by tripping the turbine and actuates passive residual heat removal to provide decay heat removal. In DCD Tier 2 Section 7.7.2.11, the applicant described the DAS as a non-safety-related system that provides a diverse backup to the protection system. The staff's safety evaluation of the AP1000 ATWS mitigation features is described in Section 7.7.2, of this report.

The applicant addressed quality assurance requirements for the SSCs providing ATWS mitigation under the regulatory treatment for non-safety systems (RTNSS) process described in SECY 95-132. WCAP-15985 provided the proposed resolution for the AP1000 RTNSS policy issue. WCAP-15985 states that the DAS functions and the associated non-class 1E DC and UPS system power supplies, are needed to meet the requirements of 10 CFR 50.62, and DAS needs to meet Generic Letter 85-06. However, WCAP-15985 did not include that GL 85-06 is also applicable to the non class 1E and UPS power systems that support the DAS ATWS functions. Therefore, the staff determined that the applicant should clearly state the quality assurance requirements that are applicable to the DAS and non-class 1E and UPS systems for the purposes of satisfying the requirements of GL 85-06. This issue is identified as Open Item 20.7-2.

Westinghouse Response:

Westinghouse agrees that the quality assurance requirements of Generic Letter 85-06 are applicable to both the DAS and the non-Class 1E dc and UPS system. The DCD and WCAP-15985 will be revised as shown below.

Design Control Document (DCD) Revision:

DCD Section 8.3.2.1.2 will be revised as shown:

8.3.2.1.2 Non-Class 1E DC and UPS System

The non-Class 1E dc and UPS system consists of the electric power supply and distribution equipment that provide dc and uninterruptible ac power to the plant non-Class 1E dc and ac loads that are critical for plant operation and investment protection and to the hydrogen igniters located inside containment. The non-class 1E dc and UPS system is comprised of two subsystems representing two separate power supply trains. The subsystems are located in separate rooms in the annex building. Figure 8.3.2-3, non-Class 1E dc and UPS system one line diagram represents the distribution configuration.

Each subsystem consists of separate dc distribution buses. These two buses can be connected by a normally open circuit breaker to enhance the power supply source availability.

Each dc subsystem includes battery chargers, stationary batteries, dc distribution equipment, and associated monitoring and protection devices.

DC buses 1, 2, and 3 (See Figure 8.3.2-3) provide 125 Vdc power to the associated inverter units that supply the ac power to the non-Class 1E uninterruptible power supply ac system. An alternate regulated ac power source for the UPS buses is supplied from the associated regulating transformers. DC bus 4 supplies large dc motors and other dc panel loads but not inverter loads. This configuration helps prevent the large motor starting disturbances affecting the sensitive electronics equipment fed from the inverters.

The onsite standby diesel generator backed 480 Vac distribution system provides the normal ac power to the battery chargers. Industry standard stationary batteries that are similar to the Class 1E design are provided to supply the dc power source in case the battery chargers fail to supply the dc distribution bus system loads. The batteries are sized to supply the system loads for a period of at least two hours after loss of all ac power sources.

The dc distribution switchboard houses the dc feeder protection device, dc bus ground fault detection, and appropriate metering. The component design and the current interrupting device selection follows the circuit coordination principles.

The non-Class 1E dc and UPS system is designed to meet the quality guidelines established by Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment that is not Safety-Related."

Each non-Class 1E dc distribution subsystem bus has provisions to allow the connection of a spare non-Class 1E battery charger should its non-Class 1E battery charger be unavailable due to maintenance, testing, or failure.

The non-Class 1E dc system uses the Class 1E spare battery bank (Figure 8.3.2-1) as a temporary replacement for any primary non-Class 1E battery bank. In this design configuration, the spare Class 1E battery bank would be connected to the non-Class 1E dc bus but could not simultaneously supply Class 1E safety loads nor perform safety-related functions. Additionally, the design includes two current interrupting devices placed in series with the main feed from the spare battery that are fault-current activated. This will preserve the spare Class 1E battery integrity should the non-Class 1E bus experience an electrical fault. This arrangement will not degrade the electrical independence of the Class 1E safety circuits.

PRA Revision:

None

WCAP Revision:

WCAP-15985 will be revised as shown:

10.3.1 Instrumentation Systems

The instrumentation systems are as follows:

- DAS (PRA Event Mitigation)

  A description of the following DAS manual controls is included in DCD subsection 7.7.1.11:

  – Reactor trip
  – PRHR HX and IRWST gutter valves (AOV)
  – CMT isolation valves (AOV)
  – ADS stages 1, 2, 3 (MOV), and stage 4 (squib)
  – IRWST injection isolation valves (squib)
  – Containment recirculation isolation valves (squib)
  – PCS water drain valves (AOV and MOV)
  – Containment isolation valves (AOV)
  – Hydrogen ignitors

  The AP1000 D-RAP includes the DAS in DCD Table 17.4-1. The inspection, tests, analyses, and acceptance criteria (ITAACs) are provided in subsection 2.5.1.

  ~~The quality assurance guidance provided in Generic Letter 85-06 is applicable to the DAS~~.

  Table 10-3 provides recommendations for DAS Technical Specifications covering these manual controls.

- DAS (ATWS)

  A description of the DAS is included in DCD subsection 7.7.1.11. The AP1000 D-RAP includes the DAS in DCD Table 17.4-1. ITAACs are provided in subsection 2.5.1.

The quality assurance guidance provided in Generic Letter 85-06 is applicable to the DAS **because of the ATWS mitigation functions (DCD subsection 7.7.1.11).**

Table 10-2 (item 1.1) provides recommendations for DAS short-term availability controls covering the ATWS function.

- DAS (EFS)

  A description of the DAS is included in DCD subsection 7.7.1.11. The AP1000 D-RAP includes the DAS in DCD Table 17.4-1. ITAACs are provided in subsection 2.5.1.

  Table 10-2 (item 1.2) provides recommendations for DAS short-term availability controls covering the ESF actuation function.

10.3.3 Electrical Systems

The electrical system are as follows:

- AC power supply system

  A description of the onsite power system is included in DCD subsection 8.3. The AP1000 D-RAP includes the onsite standby power system in DCD Table 17.4-1. ITAACs are provided in subsection 2.6.4.

  Table 10-2 (item 3.1) provides recommendations for onsite power system short-term availability controls.

- AC power supplies (RCS open)

  A description of the offsite power system is included in DCD subsection 8.2. A description of the main AC power system is included in DCD subsection 8.3.1.

  Table 10-2 (item 3.2) provides recommendations for AC power supply short-term availability controls.

- AC power supply (long-term shutdown)

  The ancillary diesel generators provide power for post-accident monitoring, PCS water makeup (recirculation pumps), MCR cooling (MCR ancillary fans), and instrumentation room cooling (instrumentation room ancillary fans). A description of the ancillary diesel generators is included in DCD subsection 8.3.1. The AP1000 D-RAP includes the ancillary diesel generators in DCD Table 17.4-1. ITAACs are provided in subsection 2.6.1.

This equipment should be available following seismic and high wind events, which may make procurement of offsite equipment more difficult. Therefore, as a minimum, the supports for this equipment are Seismic Category II as shown in DCD Table 3.2-3. In addition, this equipment is located in a portion of the Annex Building that is a Seismic II structure. Features of this structure that protect the function of this equipment are designed and analyzed for Category 5 hurricanes, including the effects of sustained winds, maximum gusts, and associated wind-borne missiles (DCD subsection 8.3.1).

Table 10-2 (item 3.3) provides recommendations for the ancillary diesel generator short-term availability controls.

- AC power supply (DAS)

The non-class 1E dc and UPS system provides power to the DAS. A description of the non-class 1E dc and UPS system is included in DCD subsection 8.3.2. ITAACs are provided in subsection 2.6.2.

The quality assurance guidance provided in Generic Letter 85-06 is applicable to the non-class 1E dc and UPS system because of the ATWS mitigation functions (DCD subsection 8.3.2.1.2)..

Table 10-2 (item 3.4) provides recommendations for non-class 1E dc and UPS system short-term availability controls.