3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Westinghouse has submitted information in support of its Design Certification Amendment application that Westinghouse considers "proprietary" within the meaning of the definition provided in Title 10, Code of Federal Regulations, Section 2.390(b)(5). Westinghouse has requested that this information be withheld from public disclosure and the NRC staff agrees that the submitted information sought to be withheld contains proprietary commercial information and should be withheld from public disclosure. This Chapter of the NRC staff's evaluation contains proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information will appear as a blank space surrounded by "square brackets" as follows:

[ ]

The complete text of this Chapter, containing proprietary information can be found at Agencywide Documents Access and Management System (ADAMS) Accession Numbers ML101241063 (Chapter 3 without Sections 3.7 and 3.8) and ML102870605 (Sections 3.7 and 3.8). These documents can be accessed by those who have specific authorization to access Westinghouse proprietary information.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

Revisions 16 and 17 of the AP1000 Design Control Document (DCD) include a number of changes to Subsection 3.2.1, Tables 3.2-2 and 3.2-3 as well as related Chapter 17 changes for quality assurance (QA) requirements. The change to Subsection 3.2.1 is limited to a clarification regarding reference to Title 10, Part 50.34, of the Code of Federal Regulations (CFR) rather than 10 CFR Part 100. The change to Table 3.2-2 consists of the inclusion of notes to clarify the non-seismic (NS) classification of certain structures described in other DCD subsections. The changes to the Table 3.2-3 primarily involve the addition of components and their seismic classifications.

3.2.1.1 Evaluation

Staff reviewed Revisions 16 and 17 of the DCD according to the guidance in the NUREG-0800 Standard Review Plan (SRP) 3.2.1, "Seismic Classification" that references Regulatory Guides (RG) 1.29, 1.143, 1.151 and 1.189 for seismic classification of various structures, systems and components (SSCs). As identified in 10 CFR Part 52.47 the application is based on regulatory guide revisions that were in effect 6 months before the docket date of the initial application. An NRC audit of design specifications performed October 13-17, 2008, for risk-significant components was also considered relative to seismic classification ADAMS) Number ML092150664. Staff reviewed related technical reports (TR) and also reviewed the non-site-specific SSCs included in DCD Subsection 3.2.1 to determine if the scope was essentially complete.

Staff determined that the Subsection 3.2.1 change referencing 10 CFR 50.34 rather than 10 CFR Part 100 was acceptable, since 10 CFR 50.34 is referenced in the definition of the term safety-related in addition to 10 CFR Part 100. Both regulations provide similar acceptance criteria for off-site doses. The other DCD changes were primarily intended to resolve staff questions on the regulatory treatment of non-safety systems (RTNSS). Staff determined that the clarifying notes to Table 3.2-2 were acceptable on the basis that structures designated as NS have augmented seismic requirements described in other DCD subsections.

The staff's review of the DCD classification changes for RTNSS determined that, in general, the specific changes identified in the amendment are acceptable, but during the review of Revision 16, the staff identified several potential errors and omissions in a number of technical areas that needed clarification in the DCD. The staff reviewed Revision 17 to determine if the issues identified during the Revision 16 review could be closed. The staff's review evaluated the DCD changes to determine if it was appropriate to resolve these errors and omissions and these are discussed below under each topic. The technical review and resulting request for additional information (RAIs) are not considered to represent new NRC requirements, but are intended to clarify statements in the DCD and address omissions in the application that have not been reviewed in the design certification.

Augmented Seismic Requirements for RTNSS SSCs (RAI-SRP3.2.1-EMB2-01)

To comply with 10 CFR 50 General Design Criteria (GDC) 2, SSCs important to safety are to be designed to withstand earthquakes. RTNSS SSCs that are important to safety but not specifically considered safety-related need not be classified as seismic Category I, but do require additional seismic design considerations under the RTNSS process to enable them to withstand earthquakes and meet GDC 1. The extent that non-safety-related SSCs are seismically qualified is defined by the RTNSS process.

In DCD Revisions 16 and 17, a number of changes were made to the classification of SSCs including classification Table 3.2-3; and the changes in Revision 17 include previously omitted SSCs important to safety, such as the ancillary diesel generators and portions of the fire protection system.

The inclusion of the ancillary diesel generators reflects a Revision 16 RAI response defining additional seismic requirements for this RTNSS equipment to be located within buildings designed to Uniform Building Code (UBC) seismic requirements with additional requirements designated in some cases. DCD Subsection 8.3.1.1.3 identifies that the ancillary diesel generators and the fuel tanks are located in the portion of the Annex Building that is a seismic Category II structure. This location is acceptable because the supplemental seismic treatment does meet minimum requirements defined in the staff requirements memorandum (SRM) dated June 23, 1997, concerning SECY-96-128 for equipment needed post-72 hour be located such that there are no spatial interactions with any other non-seismic SSCs. On the basis of the SRM, no dynamic qualification of active equipment is necessary for SSCs needed for post-72 hour actions and staff considers equipment location in a seismic Category II building with seismic Category II anchorage to be acceptable. The RAI response also indicated that the seismic classification of SSCs is considered to be complete, but if design finalization identifies changes, the design change process should identify changes that would impact the detailed application of the classification to systems and components.

Although the standpipe portions of the fire protection system that are inside the reactor containment and auxiliary building are designated in DCD Table 3.2-3 as NS, comments in the table stipulate a seismic analysis consistent with American Society of Mechanical Engineers (ASME) Section III Class 3 systems. Staff finds this to be acceptable, since this meets the criteria for seismic analysis identified in SRP Section 9.5.1 and RG 1.189 for portions of fire protection systems.

It was still not clear what additional seismic requirements may apply to certain Class D systems and components. DCD Subsection 3.2.2.6 states that, in regard to Class D, the systems and components are not designed for seismic loads. For example, other than anchorage, the seismic requirements for the ancillary diesel generators and other equipment to ensure their functionality following a seismic event is not defined. Staff guidance in a memorandum dated July 18, 1994, pertaining to AP600, identified a proposed review approach for equipment designated as important by the RTNSS process. Although a dynamic qualification test may not be necessary for this equipment, the SRM identified an approach where a dynamic analysis or qualification of electrical and mechanical equipment by experience may be used on a case-by-case basis. Staff is concerned that seismic anchorage alone does not ensure functionality of electrical and mechanical equipment following a safe-shutdown earthquake (SSE), unless it is supported by an analysis or experience. This concern was identified during the Revision 16 review as OI-SRP3.2.1-EMB2-01.

In an attempt to resolve this Revision 16 OI, staff performed an on-site review to examine detailed design documents that could define the additional information for staff to reach a reasonable safety conclusion. The results of the on-site review are documented in the NRC report dated March 17, 2009 (ADAMS Accession Number ML0906402470). The applicant responded to OI-SRP3.2.1-EMB2-01 by referencing SECY- 96-128 and NUREG-1793, Subsection 22.5.6, but DCD Subsection 3.2.1 was not updated to identify the basis cited in the response. Westinghouse Electric Company (Westinghouse or the applicant) believes that the guidance in NRC SRM dated July 18, 1994, is not applicable to the AP1000 design certification review and the seismic design requirements imposed on components, identified as important by the RTNSS process, as identified in the AP1000 DCD in Table 3.2-3 and WCAP-15985, Revision 2, provide an appropriate level of seismic protection. The response further clarified that the design provides an alternate way of supporting long term operation of passive features using offsite supplied equipment that is independent of these RTNSS SSCs. Therefore, the applicant stated that there is no need to raise the level of seismic design requirements for these RTNSS SSCs to seismic Category I and concluded that the application of seismic Category II anchorages identified in DCD Table 3.2-3 will provide reasonable assurance that the SSCs identified by the RTNSS process as important for the post 72-hour operation are functional in the required time frame, even after the most limiting design basis earthquake.

Staff reviewed the basis for seismic requirements applicable to certain RTNSS SSCs cited in the response. SECY-96-128 and the associated memorandum referenced in the response is applicable to AP600 and states that the site be capable of sustaining all design basis events with onsite equipment and supplies for the long term. The equipment required after 72 hours need not be in automatic standby response mode, but must be readily available for connection and be protected from natural phenomena including seismic events (per GDC 2). Therefore, staff disagrees with the Westinghouse position that offsite equipment may be credited for

equipment needed post 72 hours. However, based on staff guidance, no dynamic qualification of this equipment is necessary and equipment is to be designed with seismic Category II anchorage and located within a seismic Category II structure.

Although the approach proposed in the SRM dated July 18, 1994, is applicable to AP600 rather than AP1000, this document proposed a review approach for RTNSS systems in passive designs where nonsafety-related systems designated to be important by the RTNSS process (IRP) are needed to perform their required function after an earthquake. For example, IRP systems and components should not be required to be classified as seismic Category I, but staff may consider the use of experience data for seismic qualification on a case-by-case basis. SRM dated June 23, 1997, regarding SECY-96-128 for AP600, clarified a staff position that post-72 hour SSCs need not be safety-related, but equipment anchorages must be consistent with the SSE design equipment anchorages of seismic Category I items and there should be no adverse interactions. Further, this memorandum clarified that no dynamic qualification of active equipment is necessary. Although operability or functionality is not entirely ensured unless either classified as seismic Category I or otherwise justified, it is reasonable to expect that seismic Category II anchorage and location within a seismic Category II structure will afford some degree of structural integrity. Therefore, staff accepts the Westinghouse position that the seismic classification is basically consistent with previous positions for AP600 documented in documents related to SECY-96-128 and NUREG-1793. As a result of this review, Open Item OI-SRP3.2.1-EMB2-01 is closed.

Scope (RAI-SRP3.2.1-EMB2-02)

During the review of Revision 16, staff was concerned that the scope of SSCs identified in DCD Subsection 3.2.1 does not appear to be complete and this was identified as an OI. In RAI-SRP3.2.1-EMB2-02 the applicant was requested to identify the seismic classification of any non-site-specific SSCs, such as the circulating water system, electrical items, and reactor vessel insulation, within the scope of the DCD that are not included in the DCD Tables.

The RAI response clarified that the Table 3.2-3 does not include information on electrical, instrumentation or architectural elements and identified that Table 3.2-2 will be revised to include seismic requirements for various structures and that Table 3.2-3 will be revised for the fire protection systems. The response also clarified that, although the design of some of the SSCs is the responsibility of the combined license (COL) applicant, the seismic categorization is provided as part of the design certification. The response identified the circulating water system (CWS) and raw water system (RWS) as NS.

The staff reviewed Revision 17 and determined that the changes do not entirely resolve the staff's concerns. Relative to completeness of scope in the application, the applicant included the omitted ancillary diesel generators and the fire protection system components in the DCD and references DCD Subsection 3.7.2.8 for seismic requirements applicable to NS structures. However, the seismic classification of the CWS and RWS identified in the RAI response is not included in the revised DCD Tables. Similarly, DCD Revision 17 does not include the seismic classification for the electrical and instrumentation components or other miscellaneous SSCs such as the reactor pressure vessel (RPV) insulation. This concern was identified during the review of Revision 16 as Open Item OI-SRP3.2.1-EMB2-02.

In an attempt to resolve this Revision 16 OI, staff performed an on-site review to examine detailed design documents that could define the additional information for staff to reach a reasonable safety conclusion. The results of the on-site review are documented in the NRC report dated March 17, 2009 (ADAMS Accession Number ML0906402470). The applicant responded to OI-SRP3.2.1-EMB2-01 by revising the DCD, referencing DCD Table 3.11-1 for seismic classification of electric and instrumentation equipment and stating that the detail for seismic classification contained within the AP1000 DCD is sufficient for design certification. The revised DCD includes RPV insulation as seismic Category II and additional components, such as valves, the secondary core support structure and components associated with the RCS.

The staff reviewed the applicant's response. The response adequately justifies that the seismic classification of electrical items is outside the scope of SRP Section 3.2.1, and the classification of these items in Table 3.11-1 as seismic Category I should be sufficient to support the seismic review of electric items addressed in Chapter 8. Although the response does not revise the DCD to include the seismic classification of all SSCs, such as piping, piping and instrumentation drawings (P&IDs), other sections of the DCD do identify seismic classification for piping systems. The seismic classification of SSCs added in Table 3.2-3 is consistent with RG 1.29 and GDC 2. Therefore, the staff concludes that, although the scope of SSCs seismically classified in Table 3.2-3 is not complete, other sections of the DCD do include the seismic classification of SSCs not included in Table 3.2-3. As a result of this review, OI-SRP3.2.1-EMB2-02 is closed and the proposed DCD revision will be tracked as **Confirmatory Item CI-SRP3.2.1-EMB2-02**.

<u>Augmented QA Requirements for Seismic Category II SSCs</u> (RAI-SRP3.2.1-EMB2-03)

In Revision 16 DCD Subsection 3.2.1.1.2 was revised to reference DCD Section 17.5 rather than 17.4 for the combined license QA requirements for seismic Category II SSCs. During the review of Revision 16, staff determined that DCD Table 3.2-3 included in Revision 16 did not identify specific augmented QA requirements that apply to seismic Category II SSCs. The staff was concerned that Section 3.2, Table 3.2-3 or Chapter 17 included in DCD Revision 16 do not adequately define specific augmented QA requirements of Appendix B for seismic Category II SSCs. It was not clear if the COL applicant is to provide these requirements for the procurement of non-site-specific SSCs. In RAI-SRP3.2.1-EMB2-03, the applicant was requested to clarify to what extent the pertinent QA requirements of Appendix B to 10 CFR Part 50 apply to non-site-specific seismic Category II SSCs and to identify the DCD subsection or other document that describes those requirements. The RAI response restated the DCD Subsection 3.2.1.1.2 statement that pertinent portions of 10 CFR 50 Appendix B apply to seismic Category II SSCs and that pertinent portions are those required to provide that unacceptable structural failure or interaction with seismic Category I items does not occur. The response further clarified that seismic Category II SSCs are covered by the same quality programs and procedures as seismic Category I and the extent of design activities are determined by the responsible engineers and are identified in the design specifications and design criteria documents.

The staff reviewed the changes included in Revision 17 and determined that neither DCD Section 3.2, Table 3.2-3 nor Section 17.5 has been revised to identify specific augmented QA requirements for seismic Category II SSCs. This concern was identified during the review of Revision 16 as Open Item OI-SRP3.2.1-EMB2-03.

In an attempt to resolve the Revision 16 OI, staff performed an on-site review to examine detailed design documents that could define the additional information for staff to reach a reasonable safety conclusion. The results of the on-site review are documented in the NRC report dated March 17, 2009 (ADAMS Accession Number ML0906402470). The applicant responded to OI-SRP3.2.1-EMB2-03 by stating that Westinghouse does not agree that specific QA requirements for seismic category II SSCs should be included in the DCD, but the DCD is to be revised to clarify that QA requirements are performed consistent with the Westinghouse quality plan as described in Section 17.3. In the response, Westinghouse identified three different approaches applied to non-safety-related SSCs that are subject to seismic requirements and stated that AP1000 seismic Category II SSCs are subject to the AP1000 quality plan as described in DCD SRP 17.3 QA requirements.

In a subsequent response to staff concerns, the applicant clarified its process to identify supplemental requirements for RTNSS SSCs and seismic Category II SSCs. The applicant stated that application of augmented quality assurance is a function of the RTNSS assessment, not the seismic categorization. The response identifies that the Design Reliability Assurance Program (D-RAP) described in DCD Subsection 17.4 does not impose augmented design or quality requirements on SSCs and that DCD Table 3.2-1 contains adequate reference to seismic Category II design and quality requirements. The response recognizes that DCD Subsection 3.2.2.6 does not specifically allow for the use of pertinent portions of 10 CFR 50 Appendix B to seismic Category II applications and proposes a DCD revision for clarity.

Although the applicant does not impose quality requirements based on the D-RAP, the staff believes that reliability depends on the design and quality of the SSC and that the purpose of the D-RAP is to ensure reliability using the design process. As stated in DCD Section 17.4, the AP1000 D-RAP is implemented as an integral part of the AP1000 design process to provide confidence that reliability is designed into the plant. SRP Section 17.4 also states that the objective of the reliability assurance program is to ensure that the reliability is properly considered and designed into the plant. Draft DC/COL-ISG-018, Interim Staff Guidance on NUREG-0800 SRP Section 17.4, "Reliability Assurance Program," further states that the purpose of the RAP is that the reactor is designed consistent with key assumptions (including reliability) and key insights. During the design certification (DC) phase, the DC applicant prepares details of the D-RAP and implements appropriate graded controls related to design activities for non-safety-related within scope SSCs. Those supplemental requirements/graded controls (special treatment) for risk-significant SSCs may include short term availability controls, design requirements, seismic requirements, inspections, maintenance, or QA controls to ensure reliability. One of the design considerations in the AP1000 D-RAP is that the design reflect the reliability values assumed in the design and probabilistic risk assessment (PRA) as part of procurement specifications. DCD Subsections 3.2.1.1.2 and 3.2.2.6 are to be revised to reference DCD 17.3 for augmented quality requirements for seismic Category II SSCs consistent with RG 1.29, without a specific reference to the D-RAP. The staff recognizes that the RTNSS process combined with the D-RAP should be used to establish reliability of risk-significant SSCs so that appropriate specific QA requirements may be established during the detailed design. Therefore, it is reasonable to expect appropriate QA requirements to be applied to risk-significant seismic Category II SSCs and these requirements are to be included in design or procurement specifications that can be verified when available. As a result,

OI-SRP3.2.1-EMB2-03 is closed and this proposed DCD revision will be tracked as **Confirmatory Item CI-SRP3.2.1-EMB2-03.**

List of SSCs Needed for Continued Plant Operation

10 CFR Part 50, Appendix S, IV (a)(2)(I) states that SSCs necessary for continued operation without undue risk to the health and safety of the public must remain functional and within applicable stress, strain, and deformation limits when subject to the effects of the Operating Basis Earthquake (OBE) ground motion. NUREG-0800, SRP Section 3.2.1 states that, if the applicant has set the OBE ground motion to the value one-third of the SSE ground motion, then the applicant should also provide a list of SSCs necessary for continued operation that must remain functional without undue risk to the health and safety of the public and within applicable stress, strain and deformation limits, during and following the OBE. AP1000 DCD Section 3.7 states that the OBE for shutdown is considered to be one-third of the SSE.

10 CFR Part 50, Appendix S, IV (a)(3) states that if vibratory ground motion exceeding that of the OBE ground motion or if significant plant damage occurs, the licensee must shut down the nuclear power plant, and that, prior to resuming operations, the licensee must demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public and the licensing basis is maintained. Identification of the necessary SSCs and inclusion of the equipment at the appropriate seismic classification level in the DCD would allow the plant to address the requirements when the need exists.

In an attempt to obtain this information, staff performed an onsite review to examine detailed design documents that could define the additional information for staff to reach a reasonable safety conclusion. The results of the onsite review are documented in the NRC report dated March 17, 2009 (ADAMS Accession Number ML0906402470).

In RAI-SRP3.2.1-EMB2-06, the applicant was requested to provide this list of SSCs necessary for continued operation or an alternative to address the requirements. The applicant was requested to clearly state in the DCD that contains the list of SSCs necessary for continued operation. This concern was identified as Open Item OI-SRP3.2.1-EMB2-06.

The applicant's response to OI-SRP3.2.1-EMB2-06 clarifies that the SSCs necessary to protect the public health and safety are the safety-related SSCs identified in Section 3.2.2 of the DCD and tabulated in DCD Table 3.2-3. The response does not address nonsafety-related SSCs that may be important to safety, such as RTNSS SSCs, but the applicant identifies that the capability of nonsafety-related SSCs to support power production following an OBE is an investment protection issue. The response further identifies that post earthquake planning is the responsibility of the operators and is not included in the design certification. The applicant proposes a revision to DCD Subsection 3.2.1.1 to add a statement regarding the safety-related SSCs in regard to 10 CFR 50 Appendix S. In response to further staff concerns relative to pre-earthquake planning and RG 1.166 applicability, the applicant revised its response to clarify that pre-earthquake planning is the responsibility of the combined license holder and that DCD Subsection 3.7.5.2 identifies a COL Information Item for post-earthquake procedures. The response stated that post-earthquake procedures will follow Electric Power Research Institute

(EPRI) guidance and it was noted that the COL applicant would be able to address RG 1.166 and the list of SSCs to be included in procedures.

The staff agrees that RG 1.166 is not applicable to the design certification and post-earthquake planning is the responsibility of the operators and not included in the design certification. Therefore, this is considered to be addressed in the procedures developed by the COL applicant. This item will be tracked as **Confirmatory Item CI-SRP3.2.1-EMB2-06** until the DCD is updated, as stated above.

3.2.1.2 Conclusion

The seismic classification of SSCs is, in general, consistent with RG 1.29, with the exceptions identified in DCD Appendix 1A.

Therefore, on the basis of its review of DCD Revision 17 included in Tier 2 Section 3.2.1, Tables 3.2-2 and 3.2-3, the staff concludes that the AP1000 safety-related SSCs, including their supports, are properly classified as seismic Category I, in accordance with Position C.1 of RG 1.29. In addition, the staff finds that DCD Tier 2 includes acceptable commitments to Positions C.2, C.3, and C.4 of RG 1.29. This constitutes an acceptable basis for satisfying, in part, the portion of GDC 2 that requires that all SSCs important to safety be designed to withstand the effects of natural phenomena, including earthquakes.

3.2.2 Quality Group Classification

Revisions 16 and 17 of the DCD include a number of changes to Subsection 3.2.2 and Table 3.2-3 related to the AP1000 classification system and to Chapter 17 for quality assurance (QA) requirements. The changes to Subsection 3.2.2 include a clarification regarding reference to 10 CFR 50.34 rather than 10 CFR Part 100 and clarifications regarding applicability of ASME Section III to pressure-retaining components. The changes to the Table 3.2-3 primarily involve the addition of components and their AP1000 classifications.

3.2.2.1 Evaluation

The staff reviewed the DCD Revisions 16 and 17 according to the guidance in NUREG-0800 SRP 3.2.2, Quality Group Classification, which references RG 1.26 for quality group classification of various SSCs. The staff review considered that additional detailed design information needed to be verified. An NRC audit of design specifications performed October 13-17, 2008, for risk-significant components was also considered relative to Quality Group classification (ADAMS Accession Number ML092150664). The staff also reviewed TR-103 and TR-106, which address various system changes that could have an impact on quality group classifications.

The staff determined that the DCD Section 3.2.2.1 change referencing 10 CFR 50.34 rather than 10 CFR Part 100 was acceptable since 10 CFR 50.34 as well as 10 CFR Part 100 are referenced in the definition of the term safety-related. Both regulations provide similar acceptance criteria for offsite doses. The other DCD changes were primarily intended to resolve staff questions on the regulatory treatment of non-safety systems (RTNSS). The staff also determined that the clarifying notes concerning applicability of ASME Section III to pressure

boundary components were acceptable with the understanding that ASME Section III also applies to supports for pressure boundary systems and components.

The staff's review of the DCD changes determined that, in general, the specific changes identified in the application are acceptable, but that several potential errors and omissions in a number of technical areas need clarification in the DCD. During the Revision 16 review, the staff prepared RAIs to resolve these errors and omissions and these are discussed below under each topic. The technical review and resulting RAIs are intended to clarify statements in the DCD and address omissions in the application.

Supplemental Requirements for Nonsafety-Related Passive SSCs Important to Safety (RAI-SRP3.2.2-EMB2-01)

During the review of Revision 16, the staff was concerned that neither DCD Subsection 3.2 nor Table 3.2-3 adequately defines specific supplemental quality standards and QA programs applied to nonsafety-related passive SSCs that are important to safety and risk-significant. In RAI-SRP3.2.2-EMB2-01 the applicant was requested to clarify what supplemental quality standards and QA program are applied to nonsafety-related passive SSCs that are important to safety.

The RAI response clarified that codes and standards for Class D systems and components provide an appropriate level of integrity and functionality. The response also stated that the PRA did not identify SSCs that need a more rigorous code or standard than those identified in the DCD to provide improved reliability.

The staff reviewed the applicant's response to RAI-SRP3.2.2-EMB2-01 and determined that the response partially resolves its concerns. Although the PRA and RTNSS process did not apparently identify any supplemental requirements for passive components, the staff is concerned that supplementation may be appropriate, especially where there is insufficient operating history. For example, where high density polyethylene (HDPE) piping is to be used for underground plant service water system (SWS) piping that is considered a risk-significant defense in depth RTNSS system, additional special treatment should be imposed on design and QA requirements to ensure its integrity consistent with the system's safety function. Special treatment is appropriate for buried non-metallic piping that does not have a sufficient operating history in similar applications where failures are possible, unless special precautions are taken during design, fabrication, installation and testing. Examples of supplementation applied to important to safety HDPE piping are addressed in ASME Code Cases and relief requests. Although the plant service water piping is not considered safety-related, it is important to safety and GDC 1 requires that, where generally recognized codes and standards are used, they shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Therefore, passive SSCs used in risk-significant RTNSS systems such as the SWS piping should be supplemented or modified accordingly. This concern was identified during the Revision 16 review as Open Item OI-SRP3.2.2-EMB2-01.

In an attempt to resolve the Revision 16 OI, the staff performed an onsite review to examine detailed design documents that could define the additional information for staff to reach a reasonable safety conclusion. The results of the on-site review are documented in the NRC report dated March 17, 2009 (ADAMS Accession Number ML0906402470).

The applicant's response to OI-SRP3.2.2-EMB2-01 clarified that, in regard to HDPE piping used in the SWS, which is identified as a RTNSS system, HDPE will only be used in flow paths that are not required to support the important-to-safety function of decay heat removal. Therefore, Westinghouse concluded that supplementation or modification to meet GDC1 is not required in the application of HDPE piping in the SWS.

The staff agrees that, if HDPE is only used in portions of the SWS that are not risk-significant, supplementation or modification to ensure reliability of HDPE need not be identified. However, the staff was concerned that supplementation or modification of other risk-significant passive SSCs has not been identified. The applicant's revised response clarified that the RTNSS process is independent of the D-RAP and the D-RAP does not impose supplementation as a requirement. However, the response identifies that RTNSS SSCs apply augmented QA in accordance with DCD Table 17-1, QA Requirements for SSCs Important to Investment Protection. These QA requirements and scope of SSCs included in the D-RAP for RTNSS SSCs are reviewed in other sections of this SER according to SRP Section 17.4 and draft DC/COL-ISG-018, Interim Staff Guidance on NUREG-0800 SRP 17.4, "Reliability Assurance Program." Although the applicant does not impose quality requirements based on the D-RAP, the staff's opinion is that reliability depends on the design and quality of the SSC and that the purpose of the D-RAP is to ensure reliability using the design process. As stated in DCD Section 17.4, the AP1000 D-RAP is implemented as an integral part of the AP1000 design process to provide confidence that reliability is designed into the plant. SRP Section 17.4 also states that the objective of the RAP is to ensure that reliability is properly considered and designed into the plant. DC/COL-ISG-018 concerning the D-RAP and implementing appropriate graded QA controls further states that the purpose of the RAP is to assure that the reactor is designed consistent with key assumptions (including reliability) and key insights. Supplemental requirements/graded controls (special treatment) for risk-significant SSCs may include short term availability controls, design requirements, seismic requirements, inspections, maintenance, or QA controls to ensure reliability.

One of the design considerations in the AP1000 D-RAP is that the design reflects the reliability values assumed in the design and PRA as part of procurement specifications. To be consistent with the ISG, the application should specify the QA controls related to DC design activities in accordance with the provisions in Part V, "Non-safety-related SSC Quality Controls," of SRP Section 17.5 for the nonsafety-related, within scope SSCs. Based on the ISG, the NRC verifies the DC applicant's D-RAP, including its implementation during the DC application phase, through the agency's safety evaluation review process, as well as audits. Therefore, the staff recognizes that the supplementation needed to ensure reliability assumed in the PRA is to be determined by the RTNSS process combined with the D-RAP and that the inspection, test, analyses, and acceptance criteria (ITAAC) in Table 3.7-3 of Tier 1 of the AP1000 DCD have been developed to allow review of this process. As a result, Open Item OI-SRP3.2.2-EMB2-01 is closed.

<u>Application of Unendorsed ANS Standard</u> (RAI-SRP3.2.2-EMB2-02)

DCD Revision 16 added ANS Standard 58.14 as a reference for safety classifications and this standard continues to be referenced in Revision 17. The staff was concerned that withdrawn and outdated American Nuclear Society (ANS) 58.14-1993 is not NRC-endorsed and cannot be

used as a basis for acceptability of classifications. In RAI-SRP3.2.2-EMB2-02, the applicant was requested to either reference an updated classification standard or adequately describe the classification criteria in the application.

In its response, the applicant clarified that the referenced documents provide background for the equipment classification, but the AP1000 classification approach does not rely on the endorsement of any particular standard as the basis of the classification approach.

The staff reviewed the applicant's response and concludes that, although the referenced classification standard is being included in the DCD, the NRC staff will not rely on this standard or other unendorsed standards as a basis for acceptability of classifications. On this basis, RAI-SRP3.2.2-EMB2-02 is closed.

Codes and Standards (RAI-SRP3.2.2-EMB2-03)

The SRM dated July 31, 1993, concerning SECY-93-087 identified that the staff will review passive plant designs using the newest codes and standards endorsed by the NRC and unapproved revisions to the codes and standards referenced in the DCD will be reviewed on a case-by-case basis. During the Revision 16 review, the staff was concerned that editions of codes and standards referenced in the DCD not otherwise endorsed by the NRC may not be current. In RAI-SRP3.2.2-EMB2-03, the applicant was requested to clarify which editions of various codes and standards are NRC-endorsed and to clarify if current editions of codes and standards will be applied to the detailed design and procurement of AP1000 SSCs so that these codes and editions may be reviewed on a case-by-case basis.

The applicant clarified that codes and standards are generally those in effect six months prior to the submittal of the application and these editions will be applied to the detailed design and procurement of AP1000 SSCs. The response identified that, in a limited number of cases, Westinghouse is updating the revisions of codes and standards and this change is to be specifically identified in a DCD revision.

The staff agreed that editions of codes and standards in effect six months prior to the application are acceptable and noted that the staff will have the opportunity to review future changes. DCD Section 3.2.6 Revision 17 made no changes to the referenced codes and standards editions and on this basis RAI-SRP3.2.2-EMB2-03 is closed.

Classification of Fire Protection System (RAI-SRP3.2.2-EMB2-04)

During the DCD Revision 16 review the staff was concerned that DCD Subsection 3.2.2.7 has been revised to identify that both Class F and G are used for Fire Protection Systems (FPS), but Table 3.2-3 does not identify FPS SSCs that are classified as Class F and G. The staff was concerned that the classification of the FPS in DCD Revision 16 was not complete and in RAI SRP3.2.2-EMB2-04 the applicant was requested to submit the classifications for the entire FPS.

In the RAI response, the applicant submitted a revised Table 3.2-3 for additional FPS piping and components. Staff concurs that inclusion of the revised DCD Table 3.2-3 represents a generally complete scope of FPS piping and components and that the classification of these as consistent with RG 1.29 and SRP 9.5.1 criteria is an acceptable regulatory basis. The classification of the

standpipe system as AP1000 Class F constructed to ANSI B31.1 and categorized as NS with a seismic analysis consistent with ASME Section III Class 3 is consistent with the guidance in SRP Section 9.5.1 and RG 1.189 (considered not applicable to AP1000) and is, therefore, an acceptable regulatory basis. Therefore, RAI-SRP3.2.2-EMB2-04 is closed.

3.2.2.2 Conclusion

On the basis of its review of the DCD Amendment Section 3.2.2, and the above discussion, the staff concludes that the NRC Quality Group (QG) classifications of the important to safety pressure-retaining fluid systems and their supports, as identified in DCD Tier 2, Tables 3.2-1 and 3.2-3, and related P&IDs in the DCD, are consistent with RG 1.26, other than exceptions identified in DCD Appendix 1A, and are acceptable. These tables and P&IDs identify major components in fluid systems (i.e., pressure vessels, heat exchangers, storage tanks, piping, pumps, valves, and applicable supports). In addition, P&IDs in the DCD identify the classification boundaries of interconnecting piping and valves. All of the above SSCs will be constructed in conformance with applicable ASME Code and industry standards. Conformance to RG 1.26 as described above and applicable ASME Codes and industry standards provide assurance that component quality will be commensurate with the importance of the safety functions of these systems. Therefore, the staff concludes that the application meets the requirements of GDC1 for quality group classifications.

3.3 Wind and Tornado Loadings

3.3.1 Summary of Technical Information

With regard to wind and tornado loads on the Seismic Category 1 Structures, the AP1000 design certification document (DCD) Revision 17 changes the shield building by reducing its height by 5 feet. As a result, the wind and tornado loads are also altered.

3.3.2 Combined License Information 3.3-1 and 3.5-1

The commitment to address combined operating and licensing information (DCD COL Information Items 3.3-1, "Wind and Tornado Site Interface Criteria," and 3.5-1, "External Missile Protection Requirements," concerning site interface criteria for wind and tornado by the COL applicant) is defined in TR-5, Revision 4 "AP1000 Wind and Tornado Site Interface Criteria" (Reference 1, Report No. APP-GW-GLR-020). Revision 17 of the DCD includes the following applicable changes:

- Evaluation of generic wind and tornado loadings on structures;
- Provision of the plant specific site plan and comparison with the typical site plan shown in Fig. 1.2-2, "Site Plan," of the DCD Section 1.2;
- Discussion of missiles produced by tornadoes and other external events; and
- Evaluation of other buildings for collapse and missile generation.

Based on the above mentioned evaluations, the applicant is to demonstrate that any exceedances or differences in the evaluation results from what is specified in the DCD will not compromise the safety of the nuclear power plant.

3.3.3 Evaluation

The shield building is a Seismic Category I structure located on the nuclear island (NI). The development of loads on the air baffle in the top portion of the shield building due to the design-basis wind and tornado is a safety concern. The methodology for load evaluation follows the AP600 approach combined with wind tunnel testing, which gives rise to the wind loads across the air baffle, assuming a constant tornado wind speed with the height of the building. This means that the total wind load on the structure increases with increasing height of the building. The proposed change to the DCD include a 5 ft (1 m-52 cm) reduction of the total height of the shield building. As a result, total wind loads applied to the building are altered. This alteration may influence important design parameters.

The staff reviewed the change with regard to the impact on the wind load to determine its acceptability. Since the wind loads are in direct proportion to the height of the structure, the total net load applied to the building will be less than before the change. This means that, for a fixed diameter, a reduction of 5 ft (1 m-52 cm) in height will result in approximately 2.5 percent reduction in the wind loads applied to the building. The outcome of this change of design is an increase in safety margin due to decreasing applied loads. Thus, the design change increases the degree of conservatism and is, therefore, acceptable. The staff concludes that the application meets the requirements of GDC 2.

3.3.4 Development of COL Information Items

The DCD Revision 17 via TR-5, Revision 4 (Reference 1) provides the detailed requirements specified in COL Information Items 3.3-1 and 3.5-1. In order to close out the COL Information Items 3.3-1 and 3.5-1, the following items must be addressed by the COL applicant:

With regard to site interface criteria for wind and tornado (Information Item 3.3-1), the DCD states:

> The site parameters wind speeds for which the AP1000 plant is designed are given in Table 2-1, "Site Parameters (Sheets 1 - 4), of the DCD. In addition, the design parameters applicable to tornado are given in DCD Section 3.3.2.1, including maximum rotational speed of 240 mph (385 km/h); max. translational speed of 60 mph (96 km/h); radius of max. rotational wind from center of tornado, 150 ft (45-3/4 m); atmospheric pressure drop of 2.0 psi (13.8 kPa) and rate of pressure change of 1.2 psi per sec (8.3 kPa per sec). Should the site parameters exceed those bounding conditions; the applicant will be required to demonstrate that the design conforms to the acceptance criteria.

DCD, Subsection 3.3.3, "Combined License Information" contains only the commitment that COL applicants referencing the AP1000 certified design will address site interface criteria for wind and tornado loadings. This change via TR-5 provides specific interface criteria, including necessary Information Items for the COL applicant. The COL Information Items include: development of site-specific parameters, verifications of bounding conditions, plant layout and site arrangement. Should the site parameters exceed those bounding conditions, the applicant will be required, either through analysis, testing or combined analysis and testing, to demonstrate that the design conforms to the acceptance criteria.

The staff reviewed the interface criteria for wind and tornado provided in TR-5 including evaluation of generic wind and tornado loadings on structures; discussion of missiles generated by tornadoes and extreme winds, and evaluation of missile generation and effects of building collapse on nuclear island (NI) structures. Examination of those criteria revealed that they are necessary and sufficient in providing appropriate input to the design of safety-related SSCs. These COL Information Items are deemed to show compliance with the Commission's regulations including GDC 2 in Appendix A to Part 50 of 10 CFR, and thus are acceptable.

With regard to tornado-initiated building collapse (Information Item 3.3-1) the DCD states:

> If the COL applicant has adjacent structures different from the typical site plan shown in Figure 1.2-2 of the DCD Section 1.2, a justification must be provided to show that they will not collapse, or their failure will not impair the structural integrity of the nuclear island safety-related structures. Now, the structures in the typical site plan have been evaluated for tornado-initiated failure or collapse. The analysis showed that they will not compromise the safety of the nuclear island structures or their seismic categories reclassified.

The staff reviewed the analysis and found it technically sound, except for one issue that requires further investigation. The radwaste building was evaluated for its potential collapse on the NI, demonstrating that it would not impair the structural integrity of the NI safety-related structures (see DCD Subsection 3.7.2.8.2, "Radwaste Building"). However, because of the addition of 3 liquid radwaste monitor tanks (see TR-116, Reference 2), which completely alters the structural dynamic characteristics of the building, it is not clear whether this conclusion is still valid. The staff reviewed the Westinghouse response to RAI-SRP3.7.2-SEB1-02, Revision 1, dated October 1, 2008 (ADAMS Accession Number ML082770219), and determined that it was not acceptable because the staff's calculation of the maximum kinetic energy calculated using Method 3 in DCD Subsection 3.7.2.8.2 (0.6E9 in-lb or 68E6 joules) far exceeded that of an automobile missile (2E7 in-lb or 2.26E6 joules) and water tank missile (3E5 in-lb or 3.4E4 joules) claimed in the response. The staff's calculation was based on the assumptions adopted by the RAI response that the mass of the radwaste building equals the mass of a single water tank (i.e., 144,781 lbs or 65,673 kg) and the velocity is 150 fps (105 mph or 168 km per hour) from 300 mph (134 m/s), which is the design-basis tornado wind speed. This concern was identified as Open Item OI-SRP3.7.2-SEB1-02.

Westinghouse's approach to resolve the concern was to show that during a design-basis tornado event, the three water tanks will remain stationary, not result in a moving missile, then evidently there would be no safety concern on the missile impact-induced damage to NI structures, and this OI could be closed. On May 13, 2010, the staff carried out an on-site audit on this report at the Westinghouse Twinbrook office. The safety analysis in report TR APP-1000-CCC-007, "Further Evaluation of Potential Tornado Missiles on Nuclear Island," Revision 0, shows that during a tornado event with a design-basis wind speed of 300 mph (134 m/s), a total force of 27 kips will be produced by the tornado, and applied at each water tank, according to the ASCE 7-98 Standard that is acceptable to NRC. Meanwhile, the six anchorage support bolts at each tank base were designed to resist a seismic force of up to 30 kips based on the Uniform Building Code. The conclusion was that because the applied tornado force on the tank is less than the resistance capability of the tank supports at the base, the tanks will remain

stationary, and not become a damaging missile. The staff reviewed the calculations, and performed an independent confirmatory analysis using a new edition of ASCE 7-05 Standard formula. The results showed that a tornado wind speed exceeding 141.2 m/s (316 mph) will break the anchor supports, resulting in high energy water tank missiles. Any wind speed higher than this limit will turn the tank into a missile, and therefore will not be acceptable. But because the design-basis tornado wind speed is only 300 mph (134 m/s) less than the limit with a safety margin of 5 percent, the water tanks will not become a moving missile. Based on the confirmatory analysis, the staff finds that the calculations provided by Westinghouse are acceptable. This Open Item OI-SRP3.7.2-SEB1-02 is thus closed.

With regard to missiles generated by external events (COL Information Item 3.5-1) the DCD states:

> The AP1000 tornado missiles used for design are defined in Table 2.2-1 of the DCD Subsection 3.5.1.4 in terms of missile type vs. energy spectrum, which is consistent with RG 1.76 (Reference 3). Other than tornado, missiles may be generated from external events such as transportation accidents or explosions. The COL applicant is responsible for identifying sources in the plant and the external events that could cause a producing missile to threaten the integrity of AP1000 safety-related SSCs. The missile energy should be compared with the Table in 3.5.1.4. If the external event missile has higher kinetic energy, the effect of the impact must be evaluated to show that it does not compromise the safety of the AP1000 safety-related structures.

By letter dated December 23, 2008 (ADAMS Accession Number ML083640472), Westinghouse responded to RAI-SRP3.3.2-SEB1-01 regarding the issue of missiles that are produced by the potential blow-off of the siding on the annex building as well as the turbine building. In its response, Westinghouse indicated that "The automobile in the missile spectrum included in the AP1000 would appear to bound the mass and energy of sheet metal siding. Also there are no safety-related structures, systems, and components outside of the Auxiliary Building and Shield Building. The walls of these buildings are reinforced concrete at least two feet thick. Tornado driven siding would not be expected to be a challenge to reinforced concrete walls." The staff notes that the construction of the shield building is not reinforced concrete and can best be described as "steel-concrete-steel modular wall construction." It is likely that the siding missile can penetrate the steel sheet of the modular wall of the shield building. The reanalysis of the shield building for a tornado-driven siding missile strike was identified as Open Item OI-SRP3.3.2-SEB1-01. An onsite audit meeting was held on February 24, 2010, at the Westinghouse Twinbrook office where the penetration issue was discussed in detail based on the principles of mechanics in the areas of indentation, penetration and fracture. By letter dated March 24, 2010, Westinghouse responded to OI-SRP3.3.2-SEB1-01 regarding the damage induced by siding missiles. In the response, it concluded that the penetration will be zero according to the basic assumptions, methodology and detailed calculations presented in APP-1000-CCC-007, Revision 0, "Further Evaluation of Potential Tornado Missiles on Nuclear Island."

An onsite review of the report was performed by the staff on May 13, 2010 at the Westinghouse Twinbrook office. The review reveals that there is a basic assumption in the analysis that all kinetic energy is converted to strain energy in the siding and the target wall or roof. The possibility of conversion to thermal energy or fracture energy is ruled out with no justification,

and the penetration issue was not addressed. However, it is well-known that when two materials are brought into contact the harder material is bound to scratch or penetrate onto the softer material even if the velocity is very slow or buckling occurs at the high speed. Thus, as long as the hardness of the siding material is slightly higher than that of the building wall or the roof, a finite amount of penetration must occur. Indeed, in the confirmatory analysis performed by the staff, it was estimated, based on the data provided by Westinghouse on the siding missile, a penetration of about one inch (2.54 cm) and 20 inches (0.5 m) will result from the impact on the steel panel and concrete roof respectively when steel siding weighing 17.2 lbs (7.8 kg) travelling at a speed of 300 mph (134 m/s) makes a corner impact on the flat object. Those penetration depths were estimated using the appropriate formula given in SRP 5.3.2 "Barrier Design Procedures." There are no data available to confirm those estimates.

However, test data provided in a similar, but less severe blast test carried out by J.R. McDonald using a timber plank missile travelling at 150 mph (67 m/s), weighing 15 lbs (6.8 kg) with a 2' by 4' (0.6 m by1.2 m) contact area showed a penetration of 5/16" (8.0 mm) for a steel panel and 6" (15.2 cm) for a concrete slab. (References: (1) J.R. McDonald, "Impact Resistance of Common Building Materials to Tornado Missiles," Journal of Wind Engineering and Industrial Aerodynamics," Vol. 36, pp717-724, 1990; (2) M.K. Singhal and J.C. Walls: "Evaluation of Wind/Tornado-Generated Missile Impact," in Table 3, ORNL Conference No. 9310102-18). Those data suggest that the penetration estimates using the SRP 3.5.3 proposed formula are reasonable.

Given the potential local damage, a study was made in the confirmatory analysis to investigate whether the structural integrity of the NI structures would be compromised. First, from the geometry of the steel siding, those penetrations will produce a thru crack of 7.6-10.2 cm (3-4 inch) long in the steel wall and up to 0.5 m (20 inches) long in the Reinforced Concrete (RC) roof. It is important to note that the NI structure is under severe loads during a tornado event. The major loadings include a tornado wind load plus huge concentrated loads applied at a building location anywhere from grade to EL 293, resulting from the impacts by automobile missile strikes coming from the nearby raised parking lots (see Section 3.5.1.4). Thus, due to the resulting large bending moment created by the tornado loadings, tensile stress field is established in the structural components containing those flaws as the siding missile's striking site is always located on the tensile side. In the worst-case scenario when the crack happens to be located in the critical section where the tensile stress is the maximum, it is possible, according to the principle of fracture mechanics, that the crack will immediately propagate unstably if the applied stress intensity factor (which is a function of the crack size, geometry and the applied stress), exceeds the toughness resistance of the material (~50 ksi.in$^{1/2}$ or ~55 MPa.m$^{1/2}$). Eventually the crack will be arrested in the compressive stress zone. Thus, potentially a crack several feet long with noticeable opening can result as a consequence of the local impact damage from the tornado missile strikes. However, because of the large dimensions of the structures, a total collapse of the building is not likely, due to the residual strength of the components (e.g., inner steel panel of the S-C wall or intact rebars in the RC roof). The structural integrity can still be maintained.

Based on the Westinghouse assessment described above, the staff concluded that under the design-basis tornado wind loads, the structural integrity of the Seismic Category I structures will not be compromised from the siding missile strikes in compliance with GDC 2 and GDC 4 in Part 50 of 10 CFR. Therefore, OI-SRP3.3.2-SEB1-01 is closed. However, after a tornado

strike, the licensee is required to inspect and assess the damage to determine the plant's operability. If significant damage occurs (such as that described herewith), remedial measures must be taken, including a shutdown. Furthermore, prior to resuming operations, the licensee must demonstrate that no functional impairment remains to those features necessary for continued operation without undue risk to the public health and safety, and that the licensing basis is maintained.

The staff reviewed COL Information Item 3.5-1, including all possible types of missiles generated and the associated kinetic energies produced as a result of external events. Pending closure of CI-SRP3.3.2-SEB1-01, the staff determines that, in general, the kinetic energies produced falls in the scope of RG 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants," guidelines and thus conforms to GDC 4, "Environmental and Dynamic Effects Design Bases," in Appendix A to Part 50 of 10 CFR, which requires that Structures, Systems and Components (SSCs) important to safety be protected from the effects of missiles.

3.3.5 Conclusions

There are two major revisions in the DCD Section 3.3. The first change involves the design change of the shield building geometry. The shield building height was reduced by 5 feet (1-1/2 meters). As a result, the total design wind and tornado loads applied on the shield building are altered. The second change involves revision of COL Information Items 3.3-1 and 3.5-1.

The COL Information Item 3.3-1 defines site interface criteria for wind and tornado. Should the site parameters exceed the bounding conditions; the COL applicant will be required to demonstrate that the design conforms to the acceptance criteria.

The COL Information Item 3.5-1 defines acceptable missile type and energy consistent with RG 1.76. The COL applicant is responsible for identifying internal sources and external events. If the missile energy is higher than that depicted in RG 1.76, the effect of an impact must be evaluated to show that it will not impair the structural integrity of the NI safety-related structures. If significant damage occurs (such as that described herein), remedial measures must be taken, including a shutdown. Furthermore, prior to resuming operations, the COL applicant must demonstrate that no functional impairment remains to those features necessary for continued operation without undue risk to the public health and safety, and that the licensing basis is maintained.

The staff reviewed these two proposed changes to the wind and tornado loadings as documented in AP1000 DCD, Revision 16. The staff finds that these two changes do not alter the status of AP1000 wind and tornado loads with regard to meeting the applicable acceptance criteria, including the SRP guidelines. The staff also finds that the changes have been properly incorporated into the appropriate sections of the AP1000 DCD, Revision 17. On the basis that the AP1000 wind and tornado loadings continue to meet all applicable acceptance criteria, and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to Section 3.3 of the AP1000 DCD are acceptable.

3.4 External and Internal Flooding

3.4.1 Flood Protection

3.4.1.1 Protection from External Flooding

The proposed changes to the AP1000 DCD adds design features intended to prevent rainfall accumulation on the roofs of the annex, radwaste, and diesel generator buildings, increases the storage volume of one of the fire water tanks and also includes additional features to prevent or limit infiltration of groundwater into seismic Category I structures.

3.4.1.1.1 Staff Evaluation

The staff reviewed all changes related to external flood protection, Subsection 3.4.1.1.1, in the AP1000 DCD Revision 17, in accordance with SRP Section 3.4.2, "Analysis Procedures." The regulatory basis for this section is documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The staff reviewed the proposed changes to AP1000 DCD Section 3.4.1.1.1, protection from external flooding, against the applicable acceptance criteria of the SRP Section 3.4.2.

The staff reviewed the proposed changes to the roof drainage system to determine if it would impact the accumulation of water (ponding) on the roof. The applicant claimed that ponding of water on the roof is still precluded given the additional design features.

In RAI-SRP3.4.1-RHEB-01 the staff asked the applicant to discuss how the addition of parapets with weir openings to the roof drainage system would impact the potential for ponding of water on the roofs of the annex, radwaste and diesel/generator buildings. The applicant's response explained that these buildings are not Safety Related Seismic Category 1 structures and that there are no weir openings in the design. The applicant also committed to change the DCD to reflect the change. Given this information and commitment, the staff considers RAI-SRP3.4.1-RHEB-01 to be resolved. **Confirmatory Item CI-SRP3.4.1-RHEB-01** will be used to track the planned AP1000 DCD changes in Section 3.4.1.

The staff reviewed the proposed increase in storage volume in the larger firewater storage tank. The amendment seeks to increase the tank volume from 400,000 to 490,000 gallons.

In RAI-SRP3.4.1-RHEB-02, the staff asked the applicant to assess the impact of the firewater tank failure on safety-related structures, systems and components. The applicants responded in part by referring to DCD Figure 1.2-2. The applicant explained (1) the distance from the fire water tank to the auxiliary building is 320 ft and, (2) at that distance the calculated water depth would be 2.2 inches and, (3) that the base of the fire water tank is 12 inches below the nominal plant grade of 100 ft. The applicant also explained that the site shall be graded with a minimum slope of 1 percent away from the reactor buildings. The applicant also committed to change the DCD to reflect the required site grading. Based upon the depth calculation and the required slope of the site in the vicinity of the tank and NI, along with the commitment to modify the DCD, the staff considers RAI-SRP3.4.1-RHEB-02 to be resolved. **Confirmatory Item CI-SRP3.4.1-RHEB-02** will be used to track the planned AP1000 DCD changes in Section 3.4.1.

The staff also reviewed the additional features intended to prevent or limit infiltration of groundwater into seismic Category I structures. These features include embedding piping penetrations into the wall or welding the piping to a steel sleeve embedded in the wall. The

amendment also specifies that no access openings or tunnels penetrating the exterior walls of the NI are below grade and that a waterproof membrane or waterproofing system will be installed for the seismic Category I structures below grade.

3.4.1.1.2 Conclusion

The staff reviewed the applicant's proposed changes to the AP1000 external flooding as documented in DCD, Revision 17. The staff finds that the proposed changes in the case of external flooding meet the applicable acceptance criteria defined in the SRP 3.4.1. The staff also finds that the applicant has committed to incorporate the changes into the appropriate sections of the AP1000 DCD, post Revision 17. The staff finds that all of the changes to the AP1000 external flooding are acceptable because they are in compliance with GDC 2 and GDC 4 in Appendix A to Part 50 of 10 CFR, and 10 CFR 52.63 (a)(1)(vii).

3.4.1.2 Internal Flooding

3.4.1.2.1 Summary of Technical Information

In AP1000 DCD, Revision 17, Subsection 3.4.1.2.2, the applicant proposed the following changes associated with internal flooding to DCD Tier 2 of the certified design:

- The applicant proposed to modify the AP1000 DCD Section 3.4.1.2.2.1, "Reactor Coolant System Compartment" (page 3.4-7) to describe that a portion of the steam generator compartment has a low point at 24.38m (80' 0") versus the nominal elevation of 25.30m (83' 0"). The basis for this change is described in Westinghouse TR-105, "Building and Structure Configuration, Layout, and General Arrangement Design Updates," APP-GW-GLN-105, Revision 2, October 2007.

- The applicant proposed to modify the AP1000 DCD Section 3.4.1.2.2.1, "Reactor Coolant System Compartment," (page 3.4-8) to reflect the use of three redundant Class 1E flood-up level indication racks (versus the two originally in the design). The applicant stated that this change was made to assure consistency with DCD Section 6.3.7.4.4.

- The applicant proposed to modify the AP1000 DCD Section 3.4.1.2.2.2, "Auxiliary Building Level 5 - Elevation 135'-3" (page 3.4-19) to remove the discussion of the 0.57 m3 (150 gal) potable water storage (PWS) tank rupture in the main mechanical heating, ventilation, and air conditioning (HVAC) equipment rooms, which drains to the turbine building via floor drains or to the annex building via flow under the doors. This change was due to the removal of the PWS from the Westinghouse AP1000 Scope of Certification and the basis for this change is described in Westinghouse TR-124, "Removal of PWS Source and Waste Water System (WWS) Retention Basins from Westinghouse AP1000 Scope Of Certification," APP-GW-GLN-124, Revision 0, June 2007.

- The applicant proposed the following modifications to AP1000 DCD Section 3.4.1.2.2.2, "PCS Valve Room" (Section 3.4, pages 20-21):

(a) The elevation of the PCS Valve Room is changed from 87.33m (286' 6") to 86.82m (284' 10").
(b) "With the worst crack location being the 6-inch line between the valves and the flow control orifices. This leak is not isolable from the 755,000 gallon passive containment cooling system water storage tank above the valve room."
(c) "Leakage will flow down to the landing at elevation 277' 2" where the water will flow through floor drains or under doors to the upper annulus which is then discharged through redundant drains to the storm drain."

- The applicant proposed to modify the AP1000 DCD Section 3.4.1.2.2.3, "Adjacent Structures Flooding Events, Annex Building – Nonradiologically Controlled Areas" (page 3.4-22) to read: "Water accumulation at elevation 100'-0" is minimized by floor drains to the annex building sump and by flow under the access doors leading directly to the yard area." This revision eliminates reference to the flow path through the turbine building because the access door at the 100' elevation level was eliminated from the design. The basis for this change is described on page 6 of TR-105.

- The applicant proposed to modify the AP1000 DCD Section 3.4.2.2.2.3, "Adjacent Structures Flooding Events, Radwaste Building" (page 3.4-22) to read: "The potential sources of flooding in the radwaste building are the chilled water, hot water, and fire protection systems or from failure of one of the three waste monitor tanks." The basis for this change is described in Westinghouse TR-116, "Additional Liquid Radwaste Monitor Tanks and Radwaste Building Extension," APP-GW-GLN-116, Revision 0, May 2007.

- The applicant proposed editorial format changes to the AP1000 DCD Section 3.4.1.3, "Permanent Dewatering System" (page 3.4-23). These editorial changes remove references to "COL applicant items." The basis for this change is discussed in Westinghouse APP-GW-GLR-130, Revision 0, June 2007. The staff confirmed that these changes are editorial and that no further evaluation is required.

- The applicant also modified Section 4.4, TR-105, to describe structural changes performed to the auxiliary building.

3.4.1.2.2 Evaluation

The staff reviewed all changes related to the internal flooding analysis, Section 3.4.1.2, "Evaluation of Flooding Events," in the AP1000 DCD, Revision 17, in accordance with SRP Section 3.4.1, "Internal Flood Protection for Onsite Equipment Failures." The staff reviewed the proposed changes to the AP1000 DCD Section 3.4.1.2 against the applicable acceptance criteria of SRP Section 3.4.1. The following evaluation discusses the results of the staff's review.

3.4.1.2.2.1 Watertight Doors for Internal Flood Protection

In DCD Section 3.4.1.1.2, the applicant proposed a modification to state that watertight doors, in general, are not needed to protect safe shutdown components from the effects of internal floods with the exception of two watertight doors, those on the two waste holdup tank compartments. In NUREG-1793, SER Section 3.4.1.2, the staff concluded: "There are no watertight doors used for internal flood protection because they are not needed to protect safe-shutdown components from the effects of internal flooding."

In its review of DCD Section 3.4.1.1.2, the staff identified areas in which additional information was necessary to complete its evaluation of the applicant's change. In the DCD, the applicant does not describe those safety components that are protected via the added watertight doors on two waste holdup tank compartments, and does not reference a Westinghouse TR as justification. The staff requested the applicant in RAI-SRP3.4.1-SBPA-01 (ADAMS Accession Number ML081650265) to demonstrate compliance with GDC 4, "Environmental and Dynamic Effects Design Bases, by addressing the following:

1) Identify the flood source(s) associated with the spent fuel pit flooding event and the potential flood volume;

2) Provide the volume of a waste hold-up tank compartment; and

3) Identify the safe shutdown components which are protected by these watertight doors, and provide the design criteria applied for the proper functioning of these doors in the internal flood events considered.

In its July 3, 2008 response (ADAMS Accession Number ML081900159), the applicant modified the text of DCD Section 3.4.1.1.2 to reflect that the two watertight doors added during Revision 17 of the DCD were not added to protect safe-shutdown components from the effects of internal floods. These doors were added to provide additional defense-in-depth capability to retain spent fuel pool water within either a single waste holdup tank room or both waste tank rooms to limit consequences of a beyond-design-basis failure of the spent fuel pit. The applicant, in its response, also stated that the volume of a waste hold-up tank compartment is 51,900 gallons. Finally, the applicant reiterated that the watertight doors are not used to protect any safe shutdown components. These watertight doors were only added to support the beyond-design-basis accident capability. The applicant stated that the watertight doors were sized to accommodate a water pressure equivalent of 68'-0" of head, which is conservatively based on the elevation head between the maximum spent fuel pool water level and the finished floor elevation of the tank rooms. No credit is taken for the pool's level being reduced due to the pool volume required to fill the room(s).

On the basis of its evaluation of the revised DCD Section 3.4.1.1.2, the staff finds that the applicant properly identified flood sources associated with the spent fuel pit flooding event, the potential flood volume, the volume of a waste hold-up tank compartment, and the safe shutdown components that are protected by these watertight doors, and the applicant provided an adequate means of protecting safety-related equipment from the identified flood hazards. Therefore, the staff concludes that the applicant's response is acceptable and the staff's concern described in RAI-SRP3.4.1-SBPA-01 is resolved.

3.4.1.2.2.2 Building Elevation Changes

In DCD Sections 3.4.1.2.2.1 and 3.4.1.2.2.2, the applicant proposed to make design updates or design description updates to reflect that the steam generator compartment low point elevation is at 24.38m (80' 0") and the PCS valve room elevation changed from 87.33m (286' 6") to 86.82m (284' 10").

Based on its evaluation of the DCD information, the staff finds that these changes do not affect the existing SER Section 3.4.1.2 assumptions or conclusions related to internal flooding events or protection and are therefore acceptable.

3.4.1.2.2.3 Addition of a Redundant Class 1E Flood-Up Level Indication Rack

In DCD Section 3.4.1.2.2.1, the applicant proposed to modify this section to reflect the use of three (versus two) redundant Class 1E flood-up level indication racks. There are no requirements for a specified level of redundancy for these sensors. Moreover, the proposed redundancy level provides an additional layer of protection and, thus, the staff considers that the proposed design demonstrates an increase in reliability when compared to the previously approved design. In addition, the staff notes that this change does not invalidate the evaluation in NUREG-1793 Section 3.4.1.2 because there is no reference to a specific redundancy level, only that redundancy is provided.

Based on its evaluation of the DCD information, the staff concludes that this change does not affect the existing SER Section 3.4.1.2 conclusions related to internal flooding events or protection in the reactor coolant system compartment.

3.4.1.2.2.4 Deletion of PWS Tank Rupture in the DCD

In DCD Section 3.4.1.2.2.2, the applicant proposed to delete the discussion of the 0.57 m^3 (150 gal) PWS tank rupture in the main mechanical HVAC equipment rooms that drains to the turbine building via floor drains or to the annex building via flow under the doors. The applicant made this change as a consequence of removing the PWS from the Westinghouse AP1000 scope of certification. The staff evaluated this change and concludes: 1) this area does not contain equipment whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity; 2) the volume of water supplied by this tank is negligible; and 3) the volume of water from a postulated rupture of this tank or any other flooding source in this area would flow through floor drains to the turbine building or under doors leading to the annex building (which does not contain equipment required to be protected from internal flooding events).

However, since the PWS is no longer included in the scope of the design certification, the staff determined that the applicant needed to confirm that this portion of the flooding analysis remains valid, as part of the interface requirements for the site-specific PWS. The staff requested the applicant to address this requirement in RAI-SRP3.4.1-SBPA-06 (ADAMS Accession Number ML081650255).

In its response to RAI-SRP3.4.1-SBPA-06, the applicant stated that the PWS inside of the standard AP1000 plant is still included in the DCD and the design certification and the

discussion of the rupture of the 150 gallon PWS tank was inadvertently removed from the DCD. The applicant revised the text in DCD Section 3.4.1.2.2.2 for the potable water tank as follows:

> Water from fire fighting, postulated pipe or potable water storage tank (150 gallons) ruptures in the main mechanical HVAC equipment rooms drains to the turbine building via floor drains or to the annex building via flow under the doors. Therefore, no significant accumulation of water occurs in this room. Floor penetrations are sealed and a 6 inch platform is provided at the elevator and stairwell such that flooding in these rooms does not propagate to levels below.

Based on its evaluation of the revised DCD Section 3.4.1.2.2.2, the staff concludes that the change does not impact the NUREG-1793 Section 3.4.1.2 assumptions, findings, or conclusions related to internal flooding events or protection because the text was revised to match the staff accepted conclusions in DCD Revision 15. On the basis of its review, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.4.1-SBPA-06 is resolved.

3.4.1.2.2.5 Volume of PCS Water Storage Tank

In DCD Section 3.4.1.2.2.2, the applicant corrected the volume of the passive containment cooling system water storage tank above the valve room to a value of 2,858 m^3 (755,000 gal). Although the applicant did not specify the reason for this change, the staff performed its evaluation assuming it is a design change. Given that the proposed volume of water is smaller than the one previously approved, the staff concludes that its effect on the flooding analysis will be conservative.

However, the staff identified areas in which additional information was necessary to complete its evaluation. In NUREG-1793 Section 6.2.1.6 (page 6-55) the staff presumed a usable volume of 2,864.42 m^3 (756,700 gallons), which is slightly more, for passive containment heat removal. The staff requested that the applicant, in RAI-SRP3.4.1-SBPA-02 (ADAMS Accession Number ML081650265), clarify and resolve the apparent discrepancy of the volume of water in the PCS water storage tank.

In its response dated July 3, 2008, the applicant stated that it agreed with the staff conclusion that the AP1000 PCS system usable PCS tank volume of 2,864.42 m^3 (756,700 gallons) is appropriate. The indicated value will be corrected in the next version of the DCD. The applicant modified the text to read "...This leak is not isolable from the 756,700 gallon passive containment cooling system water storage tank above the valve room."

Based on its evaluation of the revised DCD Section 3.4.1.1.2 text, the staff finds that the applicant clarified the PCS water storage tank design water volume available either for passive containment cooling or as a potential internal flood source and provided an adequate means of protecting safety-related equipment from the identified flood hazards. On the basis of its review, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.4.1-SBPA-02 is acceptable.

3.4.1.2.2.6 Elimination of flow path through Turbine Building for flooding events in the Annex Building – NRCA

In DCD Section 3.4.2.2.2.3, the applicant eliminated reference to a flow path through the turbine building for flooding events in the annex building, a nonradiologically controlled area (NRCA).

The staff identified areas in which additional information was necessary to complete its evaluation of the applicant's change. In NUREG-1793 Section 3.4.1.2, page 3-21, the staff previously concluded the following:

> The mechanical equipment areas located in the NRCAs include the valve/piping penetration room (Level 3), two main steam isolation valve (MSIV) rooms, and mechanical equipment rooms (Levels 4 and 5). Flood water in these areas is routed to the turbine building or the annex building via drain lines, controlled access ways, or blowout panels which vent from the MSIV room to the turbine building.

In TR-105, the applicant did not justify the effect on the internal flooding analysis results of eliminating the route through the turbine building for flooding events. The staff requested the applicant, in RAI-SRP3.4.1-SBPA-03 (ADAMS Accession Number ML081650265) to clarify the effect of elimination of the turbine building drainage pathway on the internal flooding analysis results.

In its response dated July 3, 2008, the applicant stated that the elimination of the flow path to the turbine building at the 100'-0" level was compensated by an increase in the egress door opening to Area 4 of the annex building to match the opening previously credited to the turbine building and using the same number of alternate pathways to accommodate the flood source as previously assumed. Therefore, the applicant stated that the flood level has not been changed and remains the same as provided in Revision 15 of the DCD.

The staff identified an area in DCD Section 3.4.2.2.3 in which additional information was necessary to resolve an apparent inconsistency in the paragraph which states:

> The non-Class 1E dc and UPS system (EDS) equipment with regulatory treatment of non-safety-related systems important missions are located on elevation 100'-0" in separate battery rooms. Water in one of these rooms due to manual fire fighting in the room is collected by floor drains to the annex building sump or flows to the turbine building under doors or to the yard area through doors.

In RAI-SRP3.4.1-SBPA-04 (ADAMS Accession Number ML081650265), the staff requested the applicant to clarify the apparent discrepancy in the above paragraph. The applicant was requested to clarify whether a drainage path through the turbine building remains in the flood analysis. If there is no longer a drainage path, the applicant was asked to clarify the effect of eliminating this drainage pathway on the results of the internal flooding analysis and to verify that it does not result in any increased water level buildup that would require further evaluation.

In its response dated July 3, 2008, the applicant stated that the paragraph should have been updated consistent with the previous paragraph to reflect the elimination of the flow path to the turbine building at the 100'-0" level. The applicant corrected the paragraph in DCD Section 3.4.2.2.3 as follows:

The class 1E dc and UPS system (EDS) equipment with regulatory treatment of non-safety-related systems important missions is located on elevation 100'-0" in separate battery rooms. Water in one of these rooms due to manual fire fighting in the room is collected by floor drains to the annex building sump and by flow under the access doors leading directly to the yard area.

Based on its evaluation of the responses to RAI-SRP3.4.1-SBPA-03 and RAI-SRP3.4.1-SBPA-04 and the revised DCD Section 3.4.2.2.3 paragraph, the staff finds that the applicant justified that internal flooding analysis results were bounded by the change and provided an adequate means of protecting essential equipment from the identified flood hazards. On the basis of its review, the staff concludes that the applicant's responses are acceptable and the staff's concerns described in RAI-SRP3.4.1-SBPA-03 and RAI-SRP3.4.1-SBPA-04 are resolved.

3.4.1.2.2.7 Addition of Three Waste Monitor Tanks to Flooding Analysis

In DCD Section 3.4.2.2.2.3, the applicant included three additional potential sources of flooding, namely: "failure of one of the three waste monitor tanks." The original design included three 56.78 m^3 (15,000 gal) radwaste monitor tanks which are located in the auxiliary building. In TR-116, "Additional Liquid Radwaste Monitor Tanks and Radwaste Building Extension," APP-GW-GLN-116, May 25, 2007, the applicant added three additional 56.78 m^3 (15,000 gal) radwaste monitor tanks located in the radwaste building. The additional capacity resulted from evaluation of utility operational needs, and their addition required enlarging the building footprint of the radwaste building.

The staff finds that these changes do not affect the staff conclusions regarding flooding protection requirements in the radwaste building since this building does not house equipment required to be protected from the effects of flooding. Based on its evaluation of the DCD information, the staff concludes that the change does not impact the existing SER Section 3.4.1.2 assumptions, findings, or conclusions related to internal flooding and is acceptable.

3.4.1.2.2.8 Structural Changes Performed to the Auxiliary Building (Change 11)

In TR-105, Section 4.4, the applicant described structural changes performed to the auxiliary building. In RAI-SRP3.4.1-SBPA-05 (ADAMS Accession Number ML081650265), the staff requested that the applicant clarify if these changes had any impact on the internal flooding analysis. The applicant was requested to confirm that the auxiliary building internal flooding analysis described in DCD Section 3.4.1.2.2.2 was updated to reflect these changes or remained valid. Further, the applicant was asked to discuss how these changes affect the auxiliary building analysis with initiating events in the annex building, given that some of the proposed changes involve additional connections between the annex building and the auxiliary building.

In its response dated July 3, 2008, the applicant stated that changes described in TR-105 Section 4.4 have no impact on the internal flooding analysis as described in DCD Section 3.4.1.2.2.2 and the analysis remains valid. The applicant stated that the structural changes in connections between the annex building and auxiliary building do not have any impact on the auxiliary building flooding analysis with initiating events in the annex building because the connection points are above the elevation of the drainage paths credited for these events.

On the basis of its evaluation, the staff finds that this is a design description update change which does not impact the auxiliary building internal flooding analysis because the revised connection points are above the elevation of the drainage paths credited for these events. Therefore, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.4.1-SBPA-05 is resolved.

3.4.1.2.3 Conclusion

The staff identified acceptance criteria based on the design's meeting relevant requirements in 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena"; and in GDC 4, "Environmental and Dynamic Effects Design Bases." The staff reviewed the AP1000 internal flooding design for compliance with these requirements, as referenced in SRP Section 3.4.1, and determined that the design of the AP1000 internal flooding is acceptable because the design conforms to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 internal flooding as documented in AP1000 DCD, Revision 17. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 internal flooding to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 17. On the basis that the AP1000 internal flooding design continues to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 internal flooding are acceptable.

3.4.2 Analytical and Test Procedures

The AP1000 is designed so that the maximum hydrodynamic water forces considered due to internal flooding, external flooding, and groundwater level changes caused by extreme environmental events do not jeopardize safety of the plant or the ability to achieve and maintain safe shutdown conditions. The analytical procedures for internal flooding are described in Subsection 3.4.1.2, "Evaluation of Flooding Events," where changes were reviewed with regard to their acceptability. In this subsection, the review will be focused on changes related to external flooding events and their impacts on the structural integrity of the safety related buildings.

3.4.2.1 Summary of Technical Information

With regard to adjacent structures flooding events involving the radwaste building, the proposed change to the DCD adds one more source of potential flooding from failure of one or more of the three added waste monitor tanks in the radwaste building. The basis for this change is described in TR-116, "Additional Liquid Radwaste Monitor Tanks and Radwaste Building Extension," Westinghouse Report APP-GW-GLN-116, Revision 0, May 2007 (Reference 1).

3.4.2.2 Evaluation

The staff reviewed all changes related to the external flooding analysis, Subsection 3.4.1.1, "Flood Protection Measures for Seismic Category I Structures, Systems, and Components," in

the AP1000 DCD Revision 16, in accordance with SRP Subsection 3.4.2, "Analysis Procedures." The regulatory basis for this subsection is documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design" (Reference 2). The staff reviewed the proposed changes to AP1000 DCD Subsection 3.4.2.2 relevant to external flooding against the applicable acceptance criteria of the SRP Subsection 3.4.2. The review of the internal flooding was described in Subsection 3.4.1.2, "Internal Flooding."

The staff reviewed the change with regards to the impact on the hydrodynamic load to determine its acceptability. Since the proposed change adds three additional water tanks of 15,000 gallon (56.78 cubic meters) capacity each, collapse of the radwaste building (which is a likely scenario) will have a consequence of both internal and external flooding due to the release of large quantity of liquid from failed tanks. Since all SSCs contained in the building are non-safety related, damage by internal flooding is of no safety concern. Scenarios involving internal flooding are thus acceptable to the staff because of the evaluation contained herein. However, the release of large amounts of water from the three simultaneously failed tanks could result in external flooding to the NI structures important to safety, thereby generating extra hydrodynamic loads to the Seismic Category I structures. An analysis showing these additional loads exerted from external flooding will not impair the structural integrity of the safety-related buildings is required. The staff requested the applicant to perform such an analysis in RAI-SRP3.4.2-SEB1-01:

> The design of the radwaste building has been changed to incorporate three new additional liquid waste monitor tanks and the associated piping systems (see TR-116). Provide an analysis to show that external flooding caused by the release of the liquid from tank rupture and collapse of the radwaste building due to safe shutdown earthquake (SSE) or other extreme environmental events will not impair the structural integrity of the adjacent nuclear island (NI) structures.

The applicant responded to RAI-SRP3.4.2-SEB1-01 in a letter dated December 1, 2009 (ADAMS Accession Number ML082800327). The applicant stated that the increase in flood level would be 15 cm (6 inches) more, added to the MPF flood level due to the collapse of the 3 existing water tanks located in the auxiliary building. However, the associated extra hydrodynamic forces induced were simply stated as insignificant but not evaluated. A quantitative evaluation on the generated hydrodynamic loads showing they are insignificant on the impact to safety is needed to close this OI. This concern was identified as OI-SRP3.4.2-SEB1-01. In the response of this OI dated June 10, 2010, Westinghouse provided detailed calculations to arrive at additional water level of 6 inches (15 cm), hydrostatic pressure of 0.032 ksf (1.53 kPa), and hydrodynamic pressure of 0.45 ksf (21.6 kPa) in TR APP-1000-CCC-0007, Revision 0. The staff performed an on-site review on the report regarding the methodology, input parameters and calculation procedure, and confirmed the acceptability of the report. The results of the analysis in the report showed that additional water pressures, static as well as dynamic, and increased flood level due to the rupture of water tanks are insignificant on the impact to safety or to impair safety functions needed to be performed by the NI structures. Accordingly, the staff concludes that the change meets the relevant requirements of 10 CFR Parts 50 and 52 and GDC 2 and 4 to Appendix A of Part 50.

The staff reviewed AP1000 DCD Impact Document APP-GW-GLE-012, Revision 0, "Probable Maximum Precipitation Value Increase." On August 26, 2008 an RAI-SRP2.4-RHEB-01 was

presented to Westinghouse to clarify the maximum groundwater values. This information will affect design basis static and hydrodynamic effective loads applied to Seismic Category I structures. This concern is identified as Open Item OI-SRP2.4-RHEB-01 regarding PFM level and normal groundwater level. By letter dated September 21, 2009 the response to this OIs re-confirms the design-basis PFM at the grade 100 ft EL, and the normal groundwater level up to 98 ft EL. The surface water flooding may prevent outside access to the plant site. The AP1000 is designed to allow an isolation for a period of seven days without an increase in safety risk. Thus, the maximum design groundwater elevation is set at 98 ft EL. The staff found that the clarifications in the response to OIs are acceptable and this OI is closed. Accordingly, based on the evaluations described above, the staff concluded that the change does not significantly impact the existing SER Section 2.4 assumptions and conclusions related to changes in ground water levels or protection based on 10 CFR Parts 50 and 52 and associated acceptance criteria GDC 2 and 4 in the Appendix A to Part 50.

3.4.2.3 Conclusions

The staff reviewed the applicant's proposed changes to the AP1000 external flooding as documented in DCD, Revision 17. The staff finds that the proposed changes in the case of external flooding meet the applicable acceptance criteria defined in the SRP 3.4.2. The staff also finds that the design changes have been incorporated into the appropriate sections of the AP1000 DCD, Revision 17. Based on the evaluations performed herein, the staff finds that all of the changes to the AP1000 external flooding are acceptable because they are in compliance with the 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.

3.5 Missile Protection

3.5.1.4 Missiles Generated by Tornadoes and Extreme Winds

3.5.1.4.1 Introduction

GDC 2, in part, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety functions.

GDC 4, in part, requires that SSCs important to safety shall be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit.

With respect to protection of SSCs from missiles generated by tornadoes and extreme winds, the staff reviews the design of nuclear power facilities and considers the design to be in compliance with GDC 2 and 4 if it meets the guidance of the Positions C.1, "Design-Basis Tornado Parameters," and C.2, "Design-Basis Tornado-Generated Missile Spectrum," of RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,"

In RG 1.76, automobile missiles generated by tornadoes are considered to impact at an altitude of less than 9.14 meters (30 ft) above plant grade.

The staff reviewed the design of protection of SSCs from missiles generated by tornadoes and extreme winds for AP1000 facility. In NUREG-1793,"Final Safety Evaluation Report Related to

Certification of the AP1000 Standard Design," the staff concluded that the AP1000 design meets the requirements of GDC 2 and 4 with respect to protection against the effects of natural phenomena such as tornadoes and hurricanes and tornado generated missiles. The design also meets the guidance of RGs 1.76 with respect to the identification of missiles generated by natural phenomena. In the initial Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 COL application Final Safety Analyses Report (FSAR) Section 3.5.1.4, "Missiles Generated by Natural Phenomena," the applicant incorporated by reference Section 3.5.1.4 of the Westinghouse Electric Company AP1000 design certificate document (DCD), Revision 16, with one departure that a postulated automobile tornado missile impact is not limited to the height of 9.14 meters (30 ft) above grade on the NI. The applicant stated that the effects of a postulated automobile tornado missile impact above the height of 9.14 meters (30 ft) above grade on the NI had been evaluated by Westinghouse.

3.5.1.4.2 Technical Evaluation

During its review of Section 3.5.1.4 of the VCSNS Units 2 and 3 COL applications FSAR, the staff identified areas in which it needed additional information to complete the evaluation of the departure stated in Section 3.5.1.4 of the VCSNS Units 2 and 3 COL application FSAR. Therefore, in an RAI (RAI COL03.05.01.04-1) the staff requested the applicant to describe/provide their evaluation of the postulated automobile tornado missile striking plant structures at elevations higher than 9.14 meters (30 ft) above plant grade due to elevated local topography located within 804.67 meters (1/2 mile) of the facility. In its responses dated September 10, 2009 (ADAMS Accession Number ML09250077) and October 21, 2009 (ADAMS Accession Number ML092960452), the applicant provided Westinghouse AP1000 technical evaluation summary (TR-133), "Summary of Automobile Tornado Missile 30' above Grade," which is documented in Westinghouse AP1000 TR (APP-GW-GLR-133), Revision 0, entitled "Summary of Automobile Tornado Missile 30' above Grade," dated July 2007. The applicant stated that TR-133 envelopes the impact analysis of the automobile missile above elevation 39.63 meters (130 ft) at VCSNS.

Subsequently, Westinghouse communicated to the staff that the issue regarding the effects of a postulated tornado generated automobile missile would be addressed generically in the AP1000 DCD rather than in the VCSNS Units 2 and 3 COL application FSAR. Accordingly, in its response dated February 16, 2010 (ADAMS Accession Number ML092960452) to RAI COL03.05.01.04-1, Westinghouse stated that the postulated tornado generated automobile missile could impact the plant structures up to the junction of the outer wall of the passive containment cooling water storage tank with the roof of the shield building. Westinghouse proposed a revision to AP1000 DCD Tier 2 Section 3.5.1.4 to reflect this change and stated that the proposed change, as evaluated in TR-133, would envelop all of the referenced AP1000 sites.

On March 3, 2010, the staff conducted an audit of the automobile tornado missile calculations at the Westinghouse Twinbrook office in Rockville, Maryland. The staff issued its audit report on March 24, 2010 (ADAMS Accession Number ML100750190), which identified nine audit findings. Most of these audit findings were in the nature of requesting clarifications of discrepancies between Westinghouse AP 1000 TR-133 and the DCD and more detailed descriptions regarding the protection provided for the AP1000 facility against tornado generated automobile missiles (i.e., justification for why the passive containment cooling water tank was

excluded from the automobile missile, justification for why the y-axis label was blacked out from Figure 1 in APP-GW-GLR-133, justification for why temporary blockage of the air-inlets in the shield building was not a concern, etc.). The most significant area of concern is the evaluation of the global effect of an automobile impact on the shield building including stress.

In addition, during the structural review of TR-133, Revision 0 (ADAMS Accession Number ML093040068), the staff identified an issue related to the forcing function used in the report as an input for assessing damage due to the automobile impact in the safety analysis, and found that the report did not provide any basis or justification for the input of the forcing function used for the automobile missile impact. To address this concern, Westinghouse provided TR-133, entitled: "Nuclear Island Tornado Missile Automobile Impact above 30' " to justify the use of the forcing function. Based on the review, the staff agreed that, because of the similarity of the impact, it is appropriate to use the same forcing function to perform the damage assessment. Accordingly, Westinghouse committed to add this report as a reference in TR-133, Revision 1. On May 28, 2010, Westinghouse submitted Revision 1 of TR-133 (ADAMS Accession Number ML101530053). The staff reviewed Technical Report APP-GW-GLR-133, Revision 1 (TR-133) and confirmed that the forcing function used as a basis for the analysis was added to the report as Reference 5 in the reference list.

Also, in its letter of May 27, 2010, Westinghouse provided responses to the staff's concerns regarding the evaluation of the global effect of an automobile impact on the shield building, including stress. These staff concerns, Westinghouse's responses, and the staff's evaluation of Westinghouse's responses are described below:

In the event of an automobile missile strike on the nuclear island structures 30 ft above grade, there would be two safety concerns for the seismic Category I structures: (1) local damage; and (2) global damage. The staff reviewed the analysis of local damage in APP-1000-CCC-015, Revision 0 entitled: "Nuclear Island-Tornado Missile Automobile Impact 30' Above Grade." In the report, the applicant considered an impact area 6.6 ft by 4.3 ft by the automobile missile with a shear area 1.29' x 1.98' at the weakest location. The shear resistance of the RC wall was assessed at 112.99 psi, and the maximum shear stress induced by the impact was calculated to be 89.15 psi. Since the applied shear stress is less than the concrete wall shear resistance, Westinghouse concluded that the wall is able to resist the impact from being punched through. On this basis, the staff considers that the local damage concern at the impact spot is resolved. Another local damage concern is the crack initiation at the siding missile strike site. If the site is located at a critical section, the crack may grow unstably under the maximum stress induced by the automobile missile impact force as well as the strong tornado wind load. This safety concern was addressed in Section 3.3.4.

In addressing the global damage concern, the applicant provided a safety analysis under Audit Item 8, page 6 of 7 in its Response to RAI COL03.05.01.04-1, Revision 1, dated March 24, 2010. In the report, the possibility of failure at the connector joints of the shield building structure was considered. The analysis showed that an impact force of 770 kips from the automobile missile strike will give rise to a shearing force of 770 kips and a bending moment of 114,540 kip-ft at the RC/SC connection. The shear resistance at the weakest SS site is 23,400 kips and bending moment resistance 2,898,000 kip-ft, far exceeding the applied load exerted by the missile. This provided assurance that the connector will not fail under the automobile missile strikes.

The safety concerns of global failure due to sliding and overturning at the base were addressed in the May 13, 2010 audit. The safety analysis was provided in APP-1000-CCC-007, Revision 0 entitled: "Further Evaluation of Potential Tornado Missiles on Nuclear Island." I n the report, the resistant shear and bending moment of the building were shown to far exceed the applied shear and bending moment induced by the auto impact with a safety factor of up to 300. However, the review by the staff revealed that the analysis used an incorrect bending moment arm: the center of rotation should be at the base rather than at the connector. The analysis also failed to take the tornado load of 230 ksi into account. As a result, the safety factor was dramatically reduced to less than 30 after the corrections. Westinghouse committed to make the corrections to APP-1000-CCC-007. The staff reviewed APP-1000-CCC-007, Revision 1 and confirmed that the corrections were made.

Based on the safety analysis performed by Westinghouse against global as well as local failure due to an automobile missile strike 58 m (193 ft) above grade, the staff reviewed and accepted that assurance has been provided that the structural integrity of the NI structures will not be compromised and that the change complies with 10 CFR Part 50 Appendix A, GDC 2 and GDC 4.

In addition, in Enclosure 1 to the letter dated May 27, 2010, Westinghouse proposed to revise the first bullet under AP1000 DCD Subsection 3.5.4.1 as follows:

> A massive high-kinetic-energy missile, which deforms on impact. It is assumed to be a 4000 lbs automobile impacting the structure at normal incidence with a horizontal velocity of 105 mph or a vertical velocity of 74 mph. This missile is considered at all plant elevations up to 30 ft above grade. In addition, to consider automobiles parked within half a mile of the plant at higher elevations than the plant grade elevation; the evaluation of the automobile missile is considered at all plant elevations up to the junction of outer wall of the passive containment cooling water storage tank with the roof of the shield building. This elevation is approximately 58 m (193 ft) above grade. This evaluation bounds sites with automobiles parked within half a mile of the Shield Building and Auxiliary Building at elevations up to the equivalent of 58 m (193 ft) above grade.

Based on its review and audit of Westinghouse's responses to the above-cited RAI and Westinghouse's proposed revision to the AP1000 DCD Subsection 3.5.1.4, the staff finds that the AP1000 design continues to meet the requirements of GDC 2 and GDC 4 with respect to its ability to withstand the effects of natural phenomena and contains plant features that adequately protect against the postulated automobile tornado missile. Therefore, the staff considers its concerns described in RAI-COL03.05.01.04-01 resolved. Verification of incorporation of Westinghouse's proposed revision to the AP1000 DCD Subsection 3.5.4.1 is **CI-SRP-3.5.1.4-SBPB-02**.

3.5.1.4.3 Conclusions

The staff reviewed the applicant's proposed changes to the AP1000 postulated tornado automobile missile analysis. The staff finds that the proposed changes related to the postulated tornado-generated automobile missile analysis meet the applicable acceptance criteria defined in SRP Section 3.5.1.4. Pending the resolution of the **CI-SRP-3.5.1.4-SBPB-02**, the staff finds

that the changes related to postulated tornado automobile missiles are acceptable because they are in compliance with 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.

3.5.3 Barrier Design Procedures

3.5.3.1 Summary of Technical Information

The commitment to address in the combined license information (DCD COL Information Items 3.3-1, "Wind and Tornado Site Interface Criteria" and 3.5-1, "External Missile Protection Requirements"), onsite interface criteria for missile generation and wind and tornado loadings by the COL applicant is met in TR-5, Revision 4 (Reference 1, Report Number APP-GW-GLR-020). The proposed changes to supply the details of the Information Items are incorporated into the DCD as follows:

- Evaluation of generic wind and tornado loadings on structures,
- Provision of the plant specific site plan and comparison with the typical site plan shown in Figure 1.2-2 of the DCD Section 1.2,
- Discussion of missiles produced by tornadoes and other external events, and
- Evaluation of other buildings for collapse and missile generation.

The staff evaluations are focused on the demonstration that any exceedances or differences in the evaluation results from those specified in the DCD do not compromise the safety of the nuclear power plant.

3.5.3.2 Evaluation

The AP1000 DCD Revision 16, Tier 2, proposed closure of COL Information Items 3.3-1 and 3.5-1 in Section 3.5. In order to close out the COL Information Items, the following items must be addressed by the COL applicant:

(1) Tornado-Initiated Building Collapse (Information Item 3.3-1)

If the COL applicant has adjacent structures different from the typical site plan shown in Figure 1.2-2 of DCD Section 1.2, a justification must be provided to show that they will not collapse or that their failure will not impair the structural integrity of the NI safety-related structures. The structures in the typical site plan have now been evaluated for tornado-initiated failure or collapse. The analysis shows that they will not compromise the safety of the NI structures or result in reclassification of their seismic categories.

The staff reviewed the analysis and found that the procedure followed SRP Subsection 3.5.3, "Barrier Design Procedures," and conformed to applicable codes and RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)." This is acceptable; however, there is one issue that requires further investigation: The radwaste building was evaluated for the potential impact of its collapse on the NI structures to demonstrate that it would not impair the structural integrity of the NI safety-related structures (see DCD Subsection 3.7.2.8.2). However, because of the addition of three liquid radwaste monitor tanks (see TR-116, Reference 2), which completely alters the structural dynamic characteristics of the building, it is not clear whether this conclusion is still valid. This concern is

identified as OI-SRP3.7.2-SEB1-02. Additional information on this OI is contained in Subsection 3.3.4 herein. As discussed in Section 3.3.4, the safety concern in this OI was that, in the event of a collapse of the radwaste building during a design-basis tornado strike, the three water tanks inside the building were identified as a potential threat to safety if they were allowed to get loose to become a high energy damaging missile. In an attempt to close this OI, Westinghouse provided a safety analysis in the TR, APP-1000-CCC-007, Revision 0 titled: "Further Evaluation of Potential Tornado Missiles on Nuclear Island," showing that during a design-basis tornado event the anchor supports for the three water tanks have adequate resistant strength to prevent the tanks from breaking away to become missiles. On May 13, 2010, the staff performed an onsite review on this TR at the Westinghouse Twinbrook office. The staff conducted an independent confirmatory analysis and confirmed that so long as the tornado wind speed does not exceed 316 mph, the water tanks will not become damaging missiles. Since the design-basis tornado wind speed is set at 300 mph in the DCD Revision 17, a safety margin of 5 percent is obtained. Detailed reviewed results were discussed in Section 3.3.4. Based on the assurance provided by the TR submitted by Westinghouse, the staff finds that it is acceptable, and this OI is closed.

(2) Missiles generated by external events (Information Item 3.5-1)

The AP1000 tornado missiles used for design are defined in Table 2.2.-1 of the DCD Subsection 3.5.1.4 in terms of missile type vs. energy spectrum, which is consistent with RG 1.76 (Reference 3). Other than tornado, missiles may be generated from external events such as transportation accidents or explosions. The COL applicant is responsible for identifying sources in the plant and the external events that could produce missile(s) that threaten the integrity of AP1000 safety-related SSCs. The missile energy should be compared with the table in Subsection 3.5.1.4. If the external event missile has higher kinetic energy than that given in the table, the effect of the impact must be evaluated to show that it does not compromise the safety of the AP1000 safety-related structures.

The staff reviewed this item, and found that this extra requirement in the barrier design procedure demanded in the Information Item 3.5-1 conforms to the procedure outlined in SRP Subsection 3.5.3 and the criteria dictated by GDC 4 of Appendix A to Part 50 of 10 CFR, which require that SSCs important to safety be protected from the effects of missiles, and GDC 2 concerning the capability of the structures, shields and barriers to protect SSCs important to safety from the effects of natural phenomena. However, there is one remaining issue that requires further evaluation. The issue is related to the missiles that are produced by the potential blow-off of the siding. In the annex building as well as turbine building, metallic insulated siding is permitted to blow off during the extreme environmental event. It appears that the resulting missile in this case does not belong to any missile types listed in Table 2.2-1. Moreover, it is not clear whether the energy spectrum in the table bounds the missile energies associated with the siding-generated missiles.

By letter dated December 23, 2008 (ADAMS Accession Number ML083640472), Westinghouse responded to RAI-SRP3.3.2-SEB1-01 regarding the issue of missiles that are produced by the potential blow-off of the siding on the annex building as well as the turbine building. In its response, Westinghouse indicated that "The automobile in the missile spectrum included in the AP1000 would appear to bound the mass and energy of sheet metal siding. Also there are no safety-related structures, systems, and components outside of the Auxiliary Building and Shield

Building. The walls of these buildings are reinforced concrete at least 2 ft thick. Tornado driven siding would not be expected to be a challenge to reinforced concrete walls." The staff notes that the construction of the shield building is not reinforced concrete and can best be described as "steel-concrete-steel modular wall construction." It is likely that the siding missile can penetrate the steel sheet of the modular wall of the shield building and the RC roof. Thus, the reanalysis of the shield building for a tornado-driven siding missile was OI-SRP3.3.2-SEB1-01. By letter dated March 24, 2010, Westinghouse responded to OI-SRP3.3.2-SEB1-01 regarding the issue of damage induced by siding missiles. In the response, it is concluded that the penetration will be zero according to the basic assumptions, methodology and detailed calculations presented in the TR, APP-1000-CCC-007, Revision 0, "Further Evaluation of Potential Tornado Missiles on Nuclear Island."

An independent confirmatory analysis performed by the staff showed that for a metallic plank missile, with a mass of 17.2 lbs, flying at a velocity of 300 mph, the corner impact on the shield building could cause substantial damage in the form of major cracks several feet long and that a noticeable opening may take place. Details of the analysis are discussed in Section 3.3.4. Nevertheless, because of the large dimensions of the structures, a total collapse of the building is not likely, due to the residual strength of the components (e.g., inner steel panel of the S-C wall or intact rebar in the RC roof). Thus, the structural integrity would still be maintained.

Based on the evaluations described above, the staff concluded that, under the design-basis tornado wind loads, the structural integrity of the Seismic Category I structures will not be compromised by the siding missile strikes and that those structures are, thus, in compliance with GDC 2 and GDC 4 in Appendix A to Part 50 of 10 CFR. However, after a tornado strike, the licensee is required to inspect and assess the damage to determine the plant's operability. If significant damage occurs (such as those described herewith), remedial measures must be taken, including shutdown. Furthermore, prior to resuming operations, the licensee must demonstrate that no functional impairment remains to those features necessary for continued operation without undue risk to the public health and safety, and that the licensing basis is maintained.

3.5.3.3 Conclusions

COL Information Item 3.3-1 defines the design procedure in the case of tornado-initiated building collapse. Should the non-safety-related building collapse, the COL applicant will be required to demonstrate that the design procedure for the barriers to protect the neighboring Category I structures conforms to the acceptance criteria dictated by SRP Subsection 3.5.3 and GDC 2 and GDC 4 in Appendix A to 10 CFR Part 50.

COL Information Item 3.5-1 defines acceptable missile type and energy consistent with RG 1.76. The applicant is responsible for identifying internal sources and external events that have potential of generating hazardous missiles. If the missile energy is higher than that specified in RG 1.76, the effect of impact must be evaluated as an extra requirement in the barrier design procedure to show that it will not impair the structural integrity of the adjacent NI safety-related structures.

The staff reviewed these two changes in Subsection 3.5.4, COL Information against the SRP guidelines and acceptance criteria regarding the barrier design procedure. Based on the

discussion described above by letter dated December 23, 2008 (ADAMS Accession Number ML083640472), Westinghouse responded to RAI-SRP3.3.2-SEB1-01 regarding the issue of missiles that are produced by the potential blow-off of the siding on the annex building as well as the turbine building. In its response, Westinghouse indicated that "The automobile in the missile spectrum included in the AP1000 would appear to bound the mass and energy of sheet metal siding. Also there are no safety-related structures, systems, and components outside of the Auxiliary Building and Shield Building. The walls of these buildings are reinforced concrete at least two feet thick. Tornado driven siding would not be expected to be a challenge to reinforced concrete walls." The staff notes that the construction of the shield building is not reinforced concrete and can best be described as "steel-concrete-steel modular wall construction." It is likely that the siding missile can penetrate the steel sheet of the modular wall of the shield building and the RC roof. Thus, the reanalysis of the shield building for a tornado-driven siding missile is OI-SRP3.3.2-SEB1-01. By letter dated March 24, 2010, Westinghouse responded to OI-SRP3.3.2-SEB1-01 regarding the damage issue induced by siding missiles. In the response, it is concluded that the penetration will be zero according to the basic assumptions, methodology and detailed calculations presented in the APP-1000-CCC-007, Revision 0, "Further Evaluation of Potential Tornado Missiles on Nuclear Island."

3.6 Protection against the Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

3.6.1.1 Summary of Technical Information

Section 3.6.1, "Postulated Piping Failures in Fluid Systems Inside and Outside Containment," of the AP1000 DCD, Revision 15, was approved by staff in the certified design. In the AP1000 DCD, Revision 17, the applicant has proposed to make the following changes to this section of the certified design:

1. In DCD Section 3.6.1.1, "Design Basis," paragraph J, the applicant proposed to revise those secondary, non-safety-related components that are used to mitigate postulated line ruptures. The applicant's justification characterized this change as an editorial change that provides consistency with TR-86, "Alternate Steam and Power Conversion Design," (APP-GW-GLN-018).

2. In DCD Section 3.6.1.3.3, "Special Protection Considerations," the applicant proposed to delete the following statement in the criterion for instrumentation required to function following a pipe rupture: "In the event of a high-energy line break outside containment, the only safety-related instrumentation that could be affected is the pressure and flow instrumentation in the main steam isolation valve (MSIV) compartment conditions resulting from a 1-square-foot break from either main steam or feedwater line in the MSIV compartment as required in order to perform its safety functions." The bullet now states that instrumentation required to function following a pipe rupture is protected. The justification for this change is discussed in TR-125, "Corrections to Tier 1 ITAAC 2.2.4 and Tier 2 Section 3.6.1.3.3 and 10.3," APP-GW-GLR-125, Revision 0, May 2007.

3. In DCD Section 3.6.4.1, "Pipe Break Hazards analysis," the applicant provided COL actions that reference back to the design basis criteria in Section 3.6.1. The applicant has proposed to revise this COL item to direct the COL applicant to address the completion of the as-designed pipe break hazards analysis.

3.6.1.2 Staff Evaluation

The staff reviewed all changes to the Section 3.6.1 in the AP1000 DCD Revision 17 in accordance with SRP Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment." The regulatory basis for Section 3.6.1 of the AP1000 DCD is documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The staff reviewed the proposed changes to the AP1000 DCD Section 3.6.1 against the applicable acceptance criteria of SRP Section 3.6.1. The staff's review of DCD Section 3.6.1 was limited to postulated piping failures outside containment. The staff's evaluation of the postulated piping failures inside containment is discussed in Section 3.6.2 of this SER.

The following evaluation discusses the results of the staff's review.

3.6.1.2.1 Design Basis Assumptions

In DCD Revision 16, Tier 2, Section 3.6.1, the applicant provided the design basis and criteria for the analysis needed to demonstrate that safety-related systems are protected from pipe ruptures. This DCD section enumerates the high- and moderate-energy systems, which are potential sources of the dynamic effects associated with pipe ruptures. It also defines separation criteria.

One of the design-basis assumptions used in the dynamic effects analysis for pipe failures included the secondary components (e.g., turbine stop, moisture separator reheater stop, and turbine bypass valves). These valves are credited with mitigating the consequences of a postulated steamline break (given a single active component failure).

In its review of DCD Revision 16, Section 3.6.1, the staff identified areas in which additional information was necessary to complete its evaluation of the applicant's change. In Revision 16 to the DCD Section 3.6.1.1 to paragraph J, the applicant amended the list of secondary components to include the turbine control and stop, the turbine interceptor and reheat stop, and the turbine bypass (steam dump) valves. However, in DCD Section 3.6.1.3.3, the secondary components list consisted of the turbine stop, the moisture separator reheater stop, and the turbine bypass valves, which was inconsistent with paragraph J of DCD Section 3.6.1.1. In RAI-SRP3.6.1-SBPA-01, the staff requested the applicant to resolve the inconsistency identified between Sections 3.6.1.1 and 3.6.1.3.3.

In its response dated July 3, 2008, the applicant acknowledged the inconsistency and confirmed that the non-safety-related valves used to mitigate postulated line ruptures, given the failure of no more than one main steam isolation valve (MSIV), are:

- Turbine Control and Stop Valves

- Turbine Bypass Valves
- Moisture Separator Reheat Supply Steam Control Valves

These valves are identified in the AP1000 Technical Specification Bases (DCD Section 16.1, B3.7.2), which states that "[t]he non-safety related turbine stop or control valves, in combination with the turbine bypass, and moisture separator reheat supply steam control valves, are assumed as a backup to isolate the steam flow path given a single failure of an MSIV."

In addition, the applicant stated, that based on their review, the inconsistency was not only in Section 3.6.1.1, paragraph J and in Section 3.6.1.3.3 of the DCD, but also in Section 10.3.1.1 of the DCD.

As part of its response, the applicant provided a markup of the AP1000 DCD, Revision 16, Sections 3.6.1.1, 3.6.1.3.3, and 10.3.1.1 to rectify the inconsistencies. The staff has confirmed that the AP1000 DCD Revision 17 has included these changes.

On the basis of its review and evaluation, the staff finds that the revisions to the DCD have corrected the inconsistencies in the application; therefore, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.6.1-SBPA-01 is resolved.

3.6.1.2.2 Protection Mechanisms

In DCD Revision 16, Tier 2, Section 3.6.1, the applicant provided the measures used in the AP1000 design to protect safety-related equipment from the dynamic effects of pipe failures. These measures include physical separation of systems and components, barriers, equipment shields, and pipe whip restraints. The specific method used depends on objectives such as adequate allowance for equipment accessibility and maintenance.

Separation between redundant safety systems is the preferred method used to protect against the dynamic effects of pipe failures. Separation is achieved using the following design features:

- locating safety-related systems away from high-energy piping
- locating redundant safety systems in separate compartments
- enclosing specific components to ensure protection and redundancy
- providing drainage systems for flood control.

The staff identified an area in which additional information was necessary to complete its evaluation of the applicant's change. There was an inconsistency between TR-125 and the DCD revision that needed to be resolved. In DCD Revision 16, Section 3.6.1.3.3, the applicant provided specific protection considerations and provided the justification for revising the DCD. However, in TR-125, the applicant deleted the entire second bullet, while in Revision 16 to the DCD, the first sentence of the second bullet remained (e.g., "Instrumentation required to function following a pipe rupture is protected.") In RAI-SRP3.6.1-SBPA-02, the staff requested the applicant to resolve this inconsistency.

In its response dated July 3, 2008, the applicant stated that in developing the markup for TR-125, Revision 0, the entire second bullet of DCD Section 3.6.1.3.3 as reflected in Section 5

of TR-125, was erroneously deleted. When preparing the DCD text, however, the first sentence of the second bullet was correctly retained since it is applicable to all safety-related instrumentation located in a harsh environment.

The applicant further stated that TR-125 Section 5.0 will be revised to be consistent with DCD Section 3.6.1.3.3 Revision 16.

On the basis of its review and evaluation, the staff finds that the change to the second bullet in DCD Section 3.6.1.3.3, Revision 17, is accurate with respect to the design specifications. The proposed change ensures that all safety-related instrumentation in a harsh environment are protected from the consequences of a pipe break. Therefore, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.6.1-SBPA-02 is resolved.

3.6.1.2.3 COL Actions

In DCD Revision 17, Section 3.6.4.1, the applicant modified COL actions with respect to pipe break hazard analysis to address the completion of the as-designed pipe hazards analysis report. While this COL Information Item does not change the design basis criteria as discussed in Section 3.6.1, the modified COL Information confirms that the piping design meets the criteria provided in Section 3.6.1.3.2 (AP1000 DCD, Table 1.8-2, COL Information Item 3.6-1). The staff evaluation of the modified COL Information Item is contained in Section 3.6.2 of this SER. The staff finds that the changes to the AP1000 DCD Section 3.6.4.1 COL action are acceptable, as they relate to the protection of safety related component outside containment from the effects of a pipe break. The protection of safety related component inside containment, from the effects of a pipe break, is discussed in Section 3.6.2 of this SER.

3.6.1.3 Conclusions

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that Westinghouse's application for design certification met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In its previous evaluations of the AP1000 DCD, Section 3.6.1, "Postulated Piping Failures in Fluid Systems Inside and Outside Containment," the staff identified acceptance criteria based on the design meeting relevant requirements in 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena"; and in GDC 4, "Environmental and Dynamic Effects Design Bases." The staff reviewed the AP1000 postulated piping failures in fluid systems outside containment design for compliance with these requirements, as referenced in SRP Section 3.6.1 and determined that the design of the AP1000 postulated piping failures, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 protection of safety related component inside containment as documented in AP1000 DCD, Revision 17. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 postulated piping failures in fluid systems outside containment to meet the applicable acceptance criteria. The

staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 17. On the basis that the AP1000 postulated piping failures in fluid systems outside containment design continue to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 DCD Section 3.6.1 are acceptable.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2.1 Summary of Technical Information

AP1000 DCD Section 3.6.4.1 identifies a COL Information Item 3.6-1. The following words represent the original Combined License Information Item commitment:

> Combined License applicants referencing the AP1000 certified design will complete the final pipe whip restraint design and address as-built reconciliation of the pipe break hazards analysis in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. The as-built pipe rupture hazards analysis will be documented in an as-built Pipe Rupture Hazards Analysis Report.
>
> Subsequent to the issuance of NUREG 1793, in a letter dated January 14, 2008, APP-GW-GLR-134 through Revision 4 and AP 1000 DCD Revisions 16 and 17, Westinghouse made some DCD changes related to this COL Information Item 3.6-1.

3.6.2.2 Evaluation

The staff's review of the changes made to COL Information Item 3.6-1 are based on the pertinent information included in DCD Revisions 16 and 17, Technical Reports APP-GW-GLR-021, APP-GW-GLR-074, and APP-GW-GLR-134 through Revision 4 as well as the proposed DCD Revision 17 changes included in the Westinghouse's letter dated January 14, 2008, and December 5, 2008. In APP-GW-GLR-021 and APP-GW-GLR-074, Westinghouse proposed to modify the COL Information Item and provided a pipe rupture hazards analysis report for staff's review. Westinghouse stated that the report addressed and documented, on a generic basis, design activities required to complete COL Information Item in DCD Section 3.6.4.1 in the AP 1000 DCD. Westinghouse further stated that when the NRC review of APP-GW-GLR-074 is complete, the included activities to address the COL Information Item in Section 3.6.4.1 will be considered complete for COL applicants referencing the AP 1000 Design Certification. On the basis of its review of that report, the staff found that there were numerous areas in the report that were incomplete (e.g., ASME Class 1 piping fatigue evaluation, the complete design of the jet shields and pipe whip restraints, use of seismic response spectrum, etc.). The staff therefore, determined that the pipe rupture analysis documented in APP-GW-GLR-074 can not be considered complete and the proposed revision to the COL Information Item 3.6-1 concerning the COL applicant's responsibility was not acceptable.

Subsequently, in a letter dated January 14, 2008, Westinghouse proposed to revise AP1000 DCD Revision 16, Section 3.6.4.1 to address NRC staff's comments on the completeness of APP-GW-GLR-074. Based on its review of the information included in DCD Revisions 16 and

17 and in APP-GW-GLR-134, the staff determined that the following additional information concerning the acceptability of the proposed COL Holder Item is needed:

1a. The staff maintains that the pipe rupture hazards analysis report in APP-GW-GLR-074 is incomplete. 10 CFR 52.79(d)3 and RG 1.206 C.III.4.3 allows the applicant to propose an alternative the COL Information Item that can not be resolved completely before the issuance of a license. It requires the applicant to provide sufficient information to justify why that item can not be completed before the issuance of a license. Furthermore, it states that the applicant should provide sufficient information on this item to support the NRC licensing decision and also to propose a method for ensuring the final closure of the item including implementation schedules to allow the coordination of activities with the NRC construction inspection program following issuance of the COL. The current DCD and APP-GW-GLR-134 do not cover the level of detail described in 10 CFR 52.79(d)3 and RG 1.206 Section C.III.4.3. Westinghouse is requested to propose an alternative along with the described justification including implementation schedules to allow the coordination of activities with the NRC construction inspection program.

1b. In some of the DCD Tier 1 tables of System Based Design Description and ITAAC, the applicant includes an acceptance criterion which states that for the as-built piping, a pipe rupture hazards analysis report exists and concludes that protection from the dynamic effects of a line break is provided. It should be noted that the pipe rupture hazards analysis report is required for all the piping systems (with the exception of LBB piping) that are within the scope of SRP Section 3.6.2. The staff's concern is that the current AP1000 system based ITAAC tables do not reflect that. Westinghouse is requested to address how the system based ITAAC approach addresses all the piping systems which are within the scope of SRP 3.6.2 and are required to be included in a pipe rupture hazards analysis performed in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5.

2. In DCD Revision 16, Section 3.6.2.5 under high energy break locations, Westinghouse stated that for ASME Class 1 piping terminal end locations are determined from the piping isometric drawings. Intermediate break locations depend on the ASME Code stress report fatigue analysis results. These results are not available at design certification. For the design of the AP1000, breaks are postulated at locations typically associated with a high cumulative fatigue usage factor. Westinghouse further stated that these locations are part of the as-built reconciliation as discussed in Subsection 3.6.4.1. As discussed in this RAI question 1a, the determination of break locations is a part of the as-designed pipe rupture hazards analysis and is not part of the as-built reconciliation. Westinghouse is requested to address this concern and to revise the DCD Section 3.6.2.5 accordingly.

By a letter dated December 5, 2008, Westinghouse provided its response to the above RAIs. Based on its review of the applicant's response, the staff agreed with the applicant that the as-built reconciliation of the pipe rupture hazards analysis report is included in the ITAAC tables of the DCD which was previously reviewed and found acceptable by the staff. However, with respect to the as-designed pipe rupture hazards analysis, the staff found that the applicant has not yet adequately addressed the staff's concern relating to the completion of the as-designed pipe rupture hazards analysis report issue. Specifically, it is not clear that the as-designed pipe rupture hazards analysis report will include all piping systems within the scope of SRP Section

3.6.2 and the report will contain all the information as outlined in AP1000 DCD Subsections 3.6.1.3.2 and 3.6.2.5. Moreover, it did not clearly address the process including the milestone for the completion of the as-designed pipe rupture hazard analysis report for all piping systems within the scope of SRP Section 3.6.2. Furthermore, based on the review of the RAI response provided by some AP1000 COL applicants, the staff found that there is a difference of opinion between Westinghouse and the COL applicants as to what will be completed and, at this point, the design is not adequately addressed.

On April 9, 2009, the staff, in an AP1000 Design Centered Working Group meeting, conveyed these specific concerns to Westinghouse and AP1000 COL applicants. Subsequently, Westinghouse requested a meeting with NRC staff to discuss its plan, schedule and scope of the as-designed pipe rupture hazard analysis report. The meeting was held on May 20, 2009, at the Westinghouse Twinbrook office. During the meeting, Westinghouse indicated that it would complete an as-designed pipe rupture hazard analysis in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 for all the piping systems within the scope of SRP Sections 3.6.1 and 3.6.2 by the end of 2009 with the exception of the completion of the design for some pipe whip restraints. The remaining pipe whip restraint design would be completed by COL applicants referencing the AP1000 certified design. In addition, Westinghouse indicated that it would include all the above information in an RAI response to address the staff's concerns related to the as-designed piping rupture hazard evaluation issue. In response to Westinghouse's proposed approach, the staff indicated that it is important that all the representative AP1000 pipe whip restraint designs be completed by Westinghouse in its as-designed pipe rupture hazards analysis report. Also, Westinghouse was requested to include a discussion in its RAI response to explain what pipe whip restraints design will be completed to support staff's audit and how they are representative of the ones that will be used in the AP1000 design.

By letters dated June 30 and July 22, 2009, Westinghouse provided its response to RAI-SRP3.6.2-EMB2-01 R3, RAI-SRP3.6.4-EMB2-01 R3, and RAI-SRP3.6.2-EMB2-01 R4 respectively. Based on its review of these RAI responses, the staff found that the applicant had not clearly and adequately addressed all the issues discussed in the May 20, 2009, meeting and, for some areas, the information included in these RAI responses was different from what Westinghouse stated in that meeting.

In its response to RAI-SRP3.6.2-EMB2-01 R4, Westinghouse stated that the as-designed pipe rupture hazards analysis report, with the exception of some pipe whip restraint and jet shield designs, would be completed by December 31, 2009, and that some pipe whip restraint and jet shield designs were not expected to be completed in time to support the advanced SER with no open items. Completion of the remaining pipe whip restraint and jet shield designs will require a modified COL Information Item to be addressed in the COL applications. Westinghouse further indicated that portions of the evaluation to complete the COL Information Item might be completed during the COL application review or after the license was issued. It should be noted that during the May 20, 2009, meeting, Westinghouse indicated that to support the NRC staff's audit, it would complete an as-designed pipe rupture hazard analysis in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 for all the piping systems (including nonsafety-related piping systems, were not addressed in Westinghouse RAI responses) within the scope of SRP Sections 3.6.1 and 3.6.2, with the exception of the completion of the design for some pipe whip restraints (as opposed to pipe whip restraints and jet shields indicated in

Westinghouse RAI responses). Furthermore, based on the information included in the RAI responses, it was not clear what pipe whip restraints and jet shields design would be completed by December 31, 2009, and how they are representative of the ones that would be used in the AP1000 design. Westinghouse was, therefore, requested again to describe in detail which pipe whip restraint and jet shield designs would be completed to support staff's audit and how these completed pipe whip restraints and jet shield designs are representative of for the AP1000 design.

In its response to RAI-SRP3.6.2-EMB2-01 R4, Westinghouse also proposed some changes to DCD Subsections 3.6.2.5 and 3.6.4.1. The proposed changes did not make clear that the effects of leakage and through-wall cracks in both high and moderate energy pipes (as opposed to moderate energy pipes identified in the RAI response) are to be evaluated as part of the as-designed pipe rupture hazards analysis. It should be noted that both dynamic effects and environmental effects resulting from breaks/leakage cracks need to be evaluated for high energy pipes, while only environmental effects resulting from leakage cracks need to be evaluated for moderate energy pipes. Moreover, based on the review of the proposed DCD Subsection 3.6.4.1 changes, it appeared that the final completion of all pipe whip restraint and jet shield designs is a COL Information Item; however, it was not clearly labeled as one. Westinghouse was requested to clearly identify it as a COL Information Item or to make it an ITAAC item. This item was considered as Open Item OI-SRP3.6.2-EMB2-01.

In its response to Open Item OI-SRP3.6.2-EMB2-01, the applicant submitted a letter dated April 16, 2010. The applicant proposed that the full scope of the as-designed pipe rupture hazards analysis be addressed in COL Information Item 3.6-1. The revised COL Information Item 3.6-1 would state that COL applicants referencing the AP1000 design would complete the as-designed pipe rupture hazards analysis according to the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures and components identified (in DCD Tier 2, Table 3.6-3) to be essential targets protected by associated mitigation features would be confirmed as part of the evaluation, and updated information would be provided as appropriate. The pipe whip restraint and jet shield design included the properties and characteristics of procured components connected to the piping, components, and walls at identified break and target locations. The design would be completed prior to installation of the piping and connected components. The COL Information Item 3.6-1 would be addressed by the COL applicant in a manner that complies with NRC guidance provided in Regulatory Guide 1.215 and outlined in Appendix 14.3A of the DCD. The applicant further stated that Westinghouse would continue to work towards completion of the as-designed pipe rupture hazards analysis, and would submit a licensing topical report to the staff documenting completion of the effort and referencing the applicable design documents. The report would support the closure of the COL Information Item for the reference standard plant.

In addition, in its response to Open Item OI-SRP3.6.2-EMB2-01, the applicant also revised DCD Tier1, Table 3.3-6 Line Item 8, which requires an as-built reconciliation of the pipe rupture hazards analysis be completed prior to fuel load. The as-built reconciliation of the pipe rupture hazards analysis is to conclude that systems, structures and components identified as essential targets are protected from dynamic and environmental effects of postulated pipe ruptures.

Based on its evaluation of the above information, the staff determines that the applicant's response adequately addressed the staff's concerns described in Open Item OI-SRP3.6.2-

EMB2-01. Specifically, the proposed COL Information Item 3.6-1 and the guidance outlined in Appendix 14.3A of the DCD will ensure that the COL applicants referencing the AP1000 design will complete the as-designed pipe rupture hazards analysis report and will make it available for staff's verification in accordance with the guidance outlined in Appendix 14.3A of the DCD. In addition, the as-designed pipe rupture hazards analysis will be performed for all the piping systems within the scope of SRP Sections 3.6.1 and 3.6.2 in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. Therefore, the applicant's RAI response adequately addressed all the staff's safety questions/concerns identified in Open Item OI-SRP3.6.2-EMB2-01. In addition, the revised DCD Tier1, Table 3.3-6 Line Item 8, provides an acceptable as-built reconciliation of pipe rupture hazards analysis and will ensure that systems, structures and components identified as essential targets are protected from dynamic and environmental effects of postulated pipe ruptures. However, the applicant will need to incorporate the proposed DCD changes in future DCD revision.

Pending the staff's review of future AP 1000 DCD updates, Open Item OI-SRP3.6.2-EMB2-01 is, therefore, considered as **Confirmatory Item** CI-SRP3.6.2-EMB2-01.

3.6.4.1.2 Conclusion

Pending the satisfactory resolution of the **Confirmatory Item CI-SRP3.6.2-EMB2-01**, as discussed above, the staff concludes that the applicant's proposed changes to the COL Information Item is acceptable because the applicant has provided an acceptable alternative along with the technical justification as described in 10 CFR 52.79(d)3 and RG 1.206 C.III.4.3 regarding Combined License Information that cannot be resolved before the issuance of a license, therefore, meets applicable 10 CFR Part 52 requirements.

3.6.2.3 Conclusion

This is being tracked as Confirmatory Item **CI-SRP3.6.2-EMB2-01**; the staff concludes that the applicant's proposed changes to COL Information Item is acceptable because the applicant has provided an acceptable alternative along with the technical justification as described in 10 CFR 52.79(d)(3) and RG 1.206 C.III.4.3.

3.6.3 Leak-Before-Break

3.6.3.1 Introduction

In Revision 16 to the AP1000 DCD, Westinghouse proposed to resolve COL Information Item 3.6-2 by addressing the as-designed leak-before-break (LBB) evaluation in Report APP-GW-GLR-022. COL Information Item 3.6-2 in the Westinghouse DCD, which is also discussed in the AP1000 FSER, NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," September 2004, as Combined License Action Item 3.6.3.1-2, specifies requirements for the as-designed evaluation of LBB characteristics in AP1000 LBB piping systems. Westinghouse submitted Report APP-GW-GLR-022, "AP1000 Leak-Before-Break Evaluation of As-Designed Piping," Revision 1 (TR-8), dated July 2006, for staff review to demonstrate that it has met the requirements of COL Information Item 3.6-2. In Revision 15 to the AP1000 DCD, Section 3.6.4.2 states:

> Combined License applications referencing the AP1000 certified design will complete the leak-before-break evaluation by comparing the results of the as-designed piping stress analysis with the bounding analysis curves [BACs] documented in Appendix 3B. The Combined License applicant may perform leak-before-break evaluation for a specific location and loading for cases not covered by the bounding analysis curves. Successfully satisfying the bounding analysis curve limits in Appendix 3B may necessitate lowering the detection limit for unidentified leakage in containment from 1.9 Lpm (0.5 gpm) to 0.9 Lpm (0.25 gpm). If so, the Combined License applicant shall provide a leak detection system capable of detecting a 0.9 Lpm (0.25 gpm) leak within 1 hour and shall modify appropriate portions of the DCD including subsections 5.2.5, 3.6.3.3, 11.2.4.1, Technical Specification 3.4.7 (and Bases), Technical Specification Bases B3.4.9, and Technical Specification 3.7.8 (and Bases). The leak-before-break evaluation will be documented in a leak-before-break evaluation report.

In Revision 16 of the AP1000 DCD, Westinghouse proposed to resolve COL Information Item 3.6-2 by addressing the as-designed leak-before-break evaluation in report APP-GW-GLR-022. The revision to Section 3.6.4.2 of the DCD states:

> The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-022, and the applicable changes are incorporated into the DCD. No additional work is required by the Combined License applicant.
>
> The following words represent the original Combined License Information item commitment, which has been addressed as discussed above:
>
> Combined License applications referencing the AP1000 certified design will complete the leak-before-break evaluation by comparing the results of the as-designed piping stress analysis with the bounding analysis curves [BACs] documented in Appendix 3B. The Combined License applicant may perform leak-before-break evaluation for a specific location and loading for cases not covered by the bounding analysis curves. Successfully satisfying the bounding analysis curve limits in Appendix 3B may necessitate lowering the detection limit for unidentified leakage in containment from 1.9 Lpm (0.5 gpm) to 0.9 Lpm (0.25 gpm). If so, the Combined License holder shall provide a leak detection system capable of detecting a 0.9 Lpm (0.25 gpm) leak within 1 hour and shall modify appropriate portions of the DCD including subsections 5.2.5, 3.6.3.3, 11.2.4.1, Technical Specification 3.4.7 (and Bases), Technical Specification Bases B3.4.9, and Technical Specification 3.7.8 (and Bases). The leak-before-break evaluation will be documented in a leak-before-break evaluation report.

The scope of this evaluation does not include piping stress analysis reports whose outputs are used as inputs to this LBB evaluation.

In Revision 16 to the AP1000 DCD, Westinghouse proposed to delete COL Information Item 3.6-3 for the LBB evaluation. COL Information Item 3.6-3 in the Westinghouse DCD, which is also discussed in the AP1000 FSER, NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," September 2004, as Combined License Action Item 3.6.3.1-1, specifies requirements for the as-built evaluation of LBB characteristics in certain AP1000 piping systems. Westinghouse submitted Report APP-GW-GLR-021,

Revision 0 (TR-06), dated June 2006, for staff review to demonstrate that COL Information Item 3.6-3 may be deleted. In Revision 15, Section 3.6.4.3 to the AP1000 DCD, COL Information Item 3.6-3 states:

> Combined License applications referencing the AP1000 certified design will address: 1) verification that the as-built stresses, diameter, wall thickness, material, welding process, pressure, and temperature in the piping excluded from consideration of the dynamic effects of pipe break are bounded by the leak-before-break bounding analysis; 2) a review of the Certified Material Test Reports or Certifications from the Material Manufacturer to verify that the ASME Code, Section III strength and Charpy toughness requirements are satisfied; and 3) complete the leak-before-break evaluation by comparing the results of the final piping stress analysis with the bounding analysis curves documented in Appendix 3B. The leak-before-break evaluation will be documented in a leak-before-break evaluation report.

In Revision 16 to the AP1000 DCD, Westinghouse proposed to resolve COL Information Item 3.6-3 by deleting the text in Section 3.6.4.3. Westinghouse provided TR-6 as justification to delete COL Information Item 3.6-3.

In Revision 17 of the AP1000 DCD, Westinghouse proposed to change the composition of the Main Steam Line (MSL) piping material. Previously, in Table 3B-1 of the DCD (Revision 15), Westinghouse identified the MSL material to be utilized as ASME SA-333 Grade 6. In Revision 17 of the DCD, Westinghouse revised its DCD in Section 3.6.3 and Appendix 3B to reflect the use of ASME SA-335 Grade 11 Alloy steel. Westinghouse stated that the composition of the main steam lines was revised to minimize the potential for erosion-corrosion.

3.6.3.2 Evaluation

3.6.3.2.1 COL Information Item 3.6-2

GDC 4, "Environmental and Missile Dynamic Effects Design Bases," of Appendix A to 10 CFR Part 50 requires that structures, systems, and components important to safety shall be appropriately protected against environmental and dynamic effects. The staff reviewed changes related to this section as it relates to the leak-before-break analysis.

Westinghouse has designated TR-08 to be the "LBB Evaluation Report," as referenced in the COL Information Item. This report has reproduced, with limited modifications due to minor piping design changes, DCD BACs documented in Appendix 3B for the AP1000 LBB piping subsystems. For each AP1000 LBB piping subsystem, there is, however, extra information added to the BAC in TR-08: a point showing the normal stress (the horizontal axis) and the maximum stress (the vertical axis) based on the piping stress analysis report for the system. The normal stress is defined as the stress at the critical location of a AP1000 LBB piping subsystem due to normal loads (deadweight + pressure + thermal expansion), which are combined by the algebraic sum method. The maximum stress is defined as the stress at the critical location of a AP1000 LBB piping subsystem due to maximum loads (deadweight + pressure + thermal expansion + safe shutdown earthquake/inertia + safe shutdown earthquake/anchor motion), which are combined by the absolute sum method. The objective of this review is to verify that the stress pair (the normal stress and the maximum stress) for each

AP1000 LBB subsystem has been calculated appropriately by Westinghouse based on the piping stress report results.

An RAI was issued on August 29, 2006. A revision for one of the RAI questions was issued on September 11, 2006. RAI-TR08-001 is related to the revised BAC for the 20.3 cm (8 in) automatic depressurization system (ADS) stages 2 and 3 (upper tier) piping. RAI-TR08-002 is related to the LBB evaluation process which starts with the piping stress report results and ends with the stress pairs for all the AP1000 LBB piping subsystems. RAI-TR08-003 is related to a design change to remove the reducing tee and to add a 35.6 cm x 20.3 cm (14 in. x 8 in.) reducer in the upper tier of the ADS piping. Westinghouse provided responses to the staff RAIs in a letter dated September 29, 2006 (ADAMS Accession Number ML062760231). Since quantitative information was provided for the revised BAC requested in RAI-TR08-001, this RAI is resolved. In RAI-TR08-003 the staff requested that Westinghouse confirm the piping design changes and their effect on the corresponding BACs. In its September 29, 2006, response, Westinghouse clarified the specific changes made to the piping design and confirmed that the changes do not require additional BACs because the BACs for 15.2 cm, 20.3 cm, and 35.6 cm (6 in., 8 in., and 14 in.) piping were developed for the ADS upper tier piping, and are, thus, bounding. Therefore, RAI-TR08-003 is resolved.

RAI-TR08-002 requested additional information regarding the process of calculating the stress pair for each AP1000 LBB piping subsystem based on the corresponding piping stress report results. This involved computer software examinations, LBB calculation demonstrations, and on-site documents review. Consequently, an audit was conducted on August 29 and 30, 2006. During the audit, the staff examined line by line two post processing software designed by different Westinghouse subcontractors for LBB evaluations. In addition, the staff audited the LBB stress-pair calculations for one software application using an as-designed AP1000 automatic depressurization system (ADS) upper-tier piping and calculations for another software application using a sample passive core cooling (PXS) piping system. As a result of this audit, the staff found that the two post-processing software applications result in accurate stress pairs for the LBB evaluation, and the use of the software procedure, which does not rely on manual input of technical data, would minimize human error.

The staff's evaluation was based on the piping stress analysis results using seismic loadings associated with an AP1000 plant situated on a hard-rock site. At that time, Westinghouse was considering revising the AP1000 seismic design to include plants situated on soil sites as well. Because the seismic loadings for a plant situated on a soil site are likely to be higher than those for a plant situated on a hard-rock site, the LBB analyses for AP1000 plants situated on soil sites (or other sites other than hard-rock) would likely be affected. Thus, the staff's evaluation of the LBB analyses considered seismic loadings for hard-rock sites only. The staff confirmed that each added stress point is enveloped by the BAC curve of its piping system, indicating that all piping systems have met the requirements of COL Information Item 3.6-2. Hence, Westinghouse has demonstrated that all as-designed AP1000 LBB subsystems for plants situated on hard-rock sites meet the GDC 4 requirements for LBB applications so that the dynamic effects of postulated high-energy line pipe breaks need not be evaluated for these systems.

In addition, the proposed justification for eliminating COL Information Item 3.6-2 is based on the staff's review of Westinghouse's detailed design information that demonstrates that the LBB

calculations are bounded by the bounding analysis curves in the AP1000 DCD. The LBB as-designed analyses as described in TR-08 (APP-GW-GLR-022) are applicable to all COL applications referencing an AP1000 plant situated on a hard-rock site. The final as-built LBB analyses will be verified by the staff as part of its verification of ITAAC.

TR-08 also confirmed that the leak detection capability limit for unidentified leakage inside containment is 1.9 Lpm (0.5 gpm) as described in the DCD.

By letter dated June 20, 2008, Westinghouse addressed the LBB evaluation for AP1000 plants situated on other-than-hard-rock sites as follows:

> The other-than-hard-rock site seismic spectra are included in the piping analysis that is within the piping DAC review. The LBB evaluation results will indicate that the bounding analysis curves for piping that was evaluated for the other-than-hard-rock seismic input are acceptable and can be addressed as part of the piping DAC review.

The staff reviewed Westinghouse's response to address LBB for as-designed piping using other than hard rock site seismic spectra. Westinghouse stated that for plants situated on other-than-hard-rock-sites, the as-designed LBB analyses would be completed in conjunction with piping DAC, now a COL item (see Section 3.6.2). The NRC staff will review the final as-built LBB analyses results as part of its review of the COL item to verify that the LBB acceptance criteria are met. On the basis of its review of the AP1000 report APP-GW-GLR-022 (TR-08), the NRC staff finds that the leak-before-break analysis contained in TR-08 meets the requirements of GDC 4 and is acceptable; COL Information Item 3.6-2 is closed.

3.6.3.2.2 COL Information Item 3.6-3

GDC 4, "Environmental and Missile Dynamic Effects Design Bases," of Appendix A to 10 CFR Part 50 requires that structures, systems, and components important to safety shall be appropriately protected against environmental and dynamic effects. The staff reviewed changes related to this section as it relates to the leak-before-break analysis.

TR-06 states that the as-built evaluation of LBB characteristics will be completed after construction of the associated piping systems, as required by the ITAACs, and deletion of the COL Information Item, which requires completion of the as-built evaluation, does not alter the as-designed LBB evaluation. Since Westinghouse's justification did not address all three requirements in COL Information Item 3.6-3, the staff requested, in letter dated August 29, 2006, that Westinghouse justify the proposed deletion of this COL Information Item in accordance with the following RAI (RAI-TR06-002):

> On page 4 of the report, you propose to delete COL Information Item 3.6-3 regarding the as-built evaluation of leak-before-break piping systems. COL Information Item 3.6-3 has three elements: "1) verification that the as-built stresses, diameter, wall thickness, material, welding process, pressure, and temperature in the piping are bounded by the leak-before-break bounding analysis; 2) a review of the Certified Material Test Reports or Certifications from the Material Manufacturer to verify that the ASME Code, Section III strength and Charpy toughness requirements are satisfied; and 3) complete the leak-before-break evaluation by comparing the results of the final piping stress analysis with

the bounding analysis curves documented in Appendix 3B." Report APP-GW-GLR-022 addressed only the third requirement in COL Information Item 3.6-3, and the ITAAC regarding LBB piping systems does not specifically address the first and the second requirements. Please justify your proposed deletion of this COL Information Item by explaining how the first and second requirements (Elements 1 and 2 above) are addressed by your phrase "several ITAAC items."

Westinghouse's response (dated September 27, 2006) to RAI-TR06-002 states that the relevant ITAACs that specify the requirements for LBB evaluations are located in the DCD as Item 6 in Table 2.1.2-4 for the reactor coolant system, Item 6 in Table 2.2.3-4 for the passive core cooling system, Item 6 in Table 2.2.4-4 for the steam generator system, and Item 6 in Table 2.3.6-4 for the normal residual heat removal systems. The following is the ITAAC requirement on LBB for these systems:

6. Each of the as-built lines identified in Table x.x.x-x as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

Except for the referenced component table number, the ITAAC requirements regarding LBB evaluation are identical for all systems mentioned above. Since the above standard ITAAC requirement regarding an LBB system is not specific enough, it might not be interpreted as including the activities specified in Items 1 and 2 of COL Information Item 3.6-3 if this COL Information Item were deleted. To relieve this concern, Westinghouse modified its technical justification for TR-06 by adding the following statement in its September 27, 2006 response:

> The activities that require procurement or fabrication include verification of the stresses, diameter, wall thickness, material, welding process, pressure, and temperature of the as-built piping. The activities that require procurement or fabrication also include a review of the Certified Material Test Reports or Certifications from the material manufacturer to verify that the ASME Code, Section III strength and Charpy toughness requirements are satisfied.

The above statement in TR-06 is essentially a restatement of the first and second requirements in COL Information Item 3.6-3. The third requirement requires applicants to complete the LBB evaluation by comparing the results of the final piping stress analysis with the bounding analysis curves documented in Appendix 3B of the AP1000 DCD. To address this, a separate report, Report APP-GW-GLR-022, Revision 1 (TR-08), dated July 2006, was submitted by Westinghouse and provides an evaluation for every as-designed LBB piping. The staff has completed its evaluation of TR-08 in Section 3.6.3.1 of this supplement and finds it acceptable. Although TR-08 significantly simplifies the work related to meeting the ITAAC LBB requirements, it is not meant to replace the ITAAC activity related to LBB. When the as-built piping information becomes available after the COL phase, a final LBB evaluation will be performed by the staff in accordance with the ITAAC scope.

Therefore, the staff found that the DCD changes, as proposed by Westinghouse in TR-06, meet the requirements of GDC 4 and are acceptable. COL Information Item 3.6-3 is resolved.

3.6.3.2.3 Composition of MSL Material

GDC 4, "Environmental and Missile Dynamic Design Bases," of Appendix A to 10 CFR Part 50 requires that structures, systems, and components important to safety shall be appropriately protected against environmental and dynamic effects. GDC 4 allows the use of analyses reviewed and approved by the Commission to eliminate from the design basis the dynamic effects of postulated pipe ruptures when the analyses demonstrate that the probability of pipe rupture is extremely low. The staff reviewed the DCD Revision 17 changes in Section 3.6.3 and Appendix 3B as they relate to affecting the LBB methodology and analysis results.

The identification of SA-335 Grade 11 alloy material for the MSL is a change from the certified design (Revision 15 of the DCD), which identified the MSL material in Table 3B-1 as SA-333 Grade 6. The applicant stated that SA-335 Grade 11 was selected for the MSL material to minimize the potential for erosion-corrosion. This material contains 1-1/4 percent Chromium that is sufficient to preclude erosion-corrosion degradation in the MSL located inside containment. The staff also reviewed Appendix 3B and Figure 3B-4 in Revision 17 in which the applicant revised its LBB analysis for this material, provided a revised bounding analysis curve for the MSL, and verified that the LBB analysis for this material remained bounding for the AP1000 DCD. On this basis, the staff finds the changes to the DCD associated with the use of SA-335 Grade 11 alloy material for the MSL to be acceptable.

3.6.3.3 Conclusion

On the basis of its review of the AP1000 report APP-GW-GLR-02 (TR-06), the NRC staff finds that the proposed deletion of COL Information Item 3.6-3 meets the requirements of GDC 4 and is acceptable based on the following: (1) the first two requirements in COL Information Item 3.6-3 are preserved in TR-06, and (2) the third requirement is maintained by meeting ITAAC requirements, as described in Item 6 of Table 2.1.2-4 for the reactor coolant system, Item 6 of Table 2.2.3-4 for the passive core cooling system, Item 6 of Table 2.2.4-4 for the steam generator system, and Item 6 of Table 2.3.6-4 for the normal residual heat removal systems. Furthermore, the staff finds that the TR-06 conclusions regarding LBB characteristics in certain AP1000 piping systems are generic and are expected to apply to all COL applications referencing the AP1000 design certification. Therefore, COL Information Item 3.6-3 is deleted.

On the basis of its review of the changes in Revision 17 of the AP1000 DCD, the NRC staff finds that the leak-before-break analysis meets the requirements of GDC 4 and is acceptable.

3.7 Seismic Design

The NRC staff has conducted a detailed technical review of the seismic design and analysis of the AP1000 structures, as documented in AP1000 DCD, Revision 17 and the technical reports (TRs) discussed below. The staff used the guidance provided in Sections 3.7.1, 3.7.2, and 3.7.3 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," to conduct its review.

In September 2004, the staff issued NUREG-1793, "Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design," for the AP1000 DCD, Revision 15. In Section 3.7 of NUREG-1793, the staff concluded that the AP1000 seismic Category 1 structures located on the nuclear island (NI) were capable of withstanding the AP1000 generic

safe-shutdown earthquake (SSE) modified spectra. Regulatory Guide (RG) 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, describes how to modify the SSE spectra for design purposes. For the AP1000 design, the modified RG 1.60 spectra are anchored at 0.3g peak ground acceleration (PGA), for a hard-rock (HR) site only. An HR site is defined as having a shear wave velocity (Vs) of the supporting media ≥ 2438.4 meters per second (m/s) (8,000 feet per second (fps)). The staff also concluded that the in-structure response spectra (ISRS) were developed in accordance with staff-accepted methods described in NUREG-0800 Sections 3.7.1 and 3.7.2; and that the applicant had identified and/or implemented analytical methods for seismic system analysis and seismic subsystem analysis, consistent with NUREG-0800 Sections 3.7.2 and 3.7.3.

Subsequent to the issuance of NUREG-1793, the applicant submitted Revisions 16 and 17 to the AP1000 DCD. The applicant also submitted the following TRs:

(1) APP-GW-S2R-010, AP1000 Standard (STD) Combined License (COL) TR-03, "Extension of Nuclear Island Seismic Analyses to Soil Sites," Revisions 0 through 4. The contents of this report are summarized in the new AP1000 DCD Appendix 3G, "Nuclear Island Seismic Analyses."

(2) APP-GW-GLR-115, AP1000 STD COL TR-115, "Effect of High Frequency Seismic Content on Structures, Systems, and Components," Revisions 0 through 2. The contents of this report are summarized in the new AP1000 DCD Appendix 3I, "Evaluation for High Frequency Seismic Input."

The AP1000 seismic design changes introduced in the revised AP1000 DCD and supporting TRs are discussed in the following paragraphs:

1. *Extension of hard-rock sites to soil sites*

   The AP1000 DCD, Revision 15 only addresses the seismic design of AP1000 for an HR site. The AP1000 certified seismic design response spectra (CSDRS) for an HR site are RG 1.60 spectra anchored at 0.3g PGA, with an additional control point specified at 25 Hertz (Hz). The same CSDRS are specified in the AP1000 DCD, Revisions 16 and 17, in which the applicant introduced soil-structure interaction (SSI) analysis to evaluate the seismic response for a range of site conditions, from firm rock (FR) to soft soil (SS). For the original HR case, the applicant applies the seismic design input at the foundation elevation (El.) 18.3 meters (m) (60 feet (ft)); for the FR to SS cases, the applicant applies the seismic design input at the finished grade in the free field (El. 30.5 m (100 ft)). The applicant evaluated the structures and developed the ISRS using the enveloped response of the multiple analyses. To support the technical basis for the extension of the AP1000 design to FR and soil sites, the applicant submitted TR-03, and summarized the report in AP1000 DCD Appendix 3G. The staff's detailed evaluation of AP1000 DCD Appendix 3G and TR-03 is described in Section 3.7.2 of this SER.

2. *Use of 3-D finite element shell models*

In the AP1000 DCD, Revision 15, the applicant used three dimensional (3D) lumped mass stick models to represent the auxiliary building, containment internal structures (CISs), shield building, and steel containment. In the AP1000 DCD, Revisions 16 and 17, the applicant uses 3D finite element shell models for all NI buildings, except the steel containment. These models are used for the SSI and fixed-base seismic analyses. The detailed descriptions of the models and results of the new analyses are provided in TR-03, and summarized in AP1000 DCD Appendix 3G. The staff's detailed evaluation of these models is described in Section 3.7.2 of this SER.

3. *Effect of High Frequency Ground Motion*

The seismic analysis and design of the AP1000 plant is based on the CSDRS, which have dominant energy content in the low frequency range (2-10 Hz). However, recent probabilistic hazard-based, site-specific spectral shapes for the Central and Eastern United States (CEUS) show significant amplification above 10 Hz. This high-frequency amplification exceeds the RG 1.60 spectral amplification upon which the AP1000 CSDRS is based. The applicant has determined that for several candidate CEUS rock sites, the site-specific ground motion response spectra (GMRS) show significant increased amplitude in the high frequency range, which exceeds the CSDRS for the AP1000. The applicant has defined generic AP1000 hard rock high frequency (HRHF) spectra, which exceed the CSDRS above 15 Hz in the horizontal direction and above 20 Hz in the vertical direction. To address the exceedances, the applicant has performed an evaluation to demonstrate that, in general, the high frequency ground motion represents a lower seismic demand on AP1000 structures, systems, and components (SSCs) than the CSDRS.

The applicant compared the responses of a sample of SSCs, using both the CSDRS and the HRHF response spectra as seismic inputs. The evaluation included building structures, reactor pressure vessel internals, primary component supports, primary loop nozzles, piping, and electro-mechanical equipment. The applicant's evaluation of HRHF ground motion is described in TR-115, and briefly summarized in the new AP1000 DCD Appendix 3I. The staff's review of the applicant's evaluation of high frequency effects is described in Section 3.7.2 of this SER.

4. *Application of Incoherency Effects*

The incoherency of seismic waves has been recognized for several decades as having an effect on structures with large dimensions, separate supports, or large distances between supports (e.g., bridges). Until recently, data to support analytical models were scarce. Luco, Abrahamson, Zerva, and others, using data from surface recordings from dense arrays located in Taiwan, Japan, and California, developed coherency models to characterize local variations in free-field ground motions to analytically capture these incoherent effects sustained by structural foundations. These data were previously based on recordings at soil sites. Recently, Abrahamson (2006) extended these coherency models to include the effects at rock sites. This coherency function approximates the known changes of motion based on spatial separation and frequency and has been incorporated into several SSI analysis codes.

The incoherency of seismic waves generally results in a reduction of structural translational responses when compared with coherent seismic motion, especially in higher frequency ranges (e.g., frequencies greater than 10 Hz). For structures of large dimensions typical of nuclear power plants designs, these translational modes can be reduced due to wave scattering, but torsion and rocking modes can be induced that can result in increased response at locations remote from the center-of-mass.

The applicant has used seismic motion incoherency in its evaluation of HRHF ground motion effects on AP1000 SSCs. The staff issued interim staff guidance (ISG) in May 2008, identifying an acceptable approach to consider the effects of incoherency on the NI foundation, specifically for HRHF seismic ground motion. The staff accepted the seismic ground motion coherency function as described in an Electric Power Research Institute (EPRI) report entitled, "Hard-Rock Coherency Functions Based on the Pinyon Flat Array Data," dated July 5, 2007. The applicant indicated that its evaluation is consistent with the staff's ISG. Because this is a first-time implementation of the staff's ISG, the staff conducted independent confirmatory analysis. The staff's detailed evaluation of the applicant's use of incoherency is described in Section 3.7.2 of this SER.

3.7.1 Seismic Input

NUREG-0800 Section 3.7.1, "Seismic Design Parameters," provides guidelines for the staff to use in reviewing issues related to the development of seismic input ground motions, percentage of critical damping values, and supporting media for seismic Category I structures. The following evaluation addresses the proposed changes to the seismic design, as described in the amendment to the AP1000 design certification (DC). As such, this evaluation revises and supplements the evaluation in corresponding sections of NUREG-1793.

3.7.1.1 Design Ground Response Spectra

In AP1000 DCD Tier 1, Section 5.0, the applicant described the AP1000 CSDRS. The staff verified that the AP1000 CSDRS remain unchanged from the AP1000 DCD, Revision 15. In AP1000 DCD Tier 2, Section 3.7.1.1, the applicant indicated that the AP1000 CSDRS have been established with a PGA of 0.3g for the AP1000 design, in both the horizontal and vertical directions. The design response spectra are based on RG 1.60 with an additional control point specified at 25 Hz. The spectral amplitude at 25 Hz is 30 percent higher than the RG 1.60 spectral amplitude.

In AP1000 DCD, Tier 2, Section 2.5.2, the applicant provided a description of how the AP1000 CSDRS are compared to the site-specific GMRS. The CSDRS are compared to the site-specific GMRS at different locations depending on the site characteristics. In AP1000 DCD Section 3.7.1.1, the applicant states that the CSDRS are applied at the foundation level (El. 18.4 m (60 ft 6 in)) in the free field at HR sites and at the finished grade (El. 50.5 m (100 ft)) in the free field at FR and soil sites. Applying the design response spectra at the foundation level in the free field for the HR sites was accepted by the staff during its AP1000 DCD, Revision 15 review. With respect to the FR and soil sites, the staff finds that the applicant's approach of applying the design response spectra at the surface (in the free field) for both FR

and soil sites is acceptable, because it is in accordance with the guidance described in NUREG-0800 Section 3.7.1.

The staff noted, however, that AP1000 DCD Section 3.7.1, Revision 17, did not provide a basis for satisfying 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," which requires the horizontal component of the SSE ground motion in the free field at the foundation elevation to have a PGA of at least 0.1g and an appropriate response spectrum. To address this concern, the staff issued request for additional information (RAI)-SRP3.7.1-SEB1-18, requesting the applicant to provide free field in-column response spectra and associated PGA generated for each of the generic-site columns (FR and soil sites) considered. This was identified as Open Item OI-SRP3.7.1-SEB1-18 in the SER with open items.

In a letter dated May 14, 2010, the applicant provided the in-column response spectra at the basemat elevation for each of the generic sites, in Figure RAI-SRP3.7.1-SEB1-18-1, attached to the response. The horizontal PGA at the basemat elevation is above 0.1g for all generic sites. On this basis, the staff determined that the requirements of 10 CFR Part 50, Appendix S, are satisfied; therefore, RAI-SRP3.7.1-SEB1-18 and the associated open item are resolved.

3.7.1.2 Critical Damping Values

In AP1000 DCD, Tier 2, Section 3.7.1.3, the applicant described the critical damping values assigned to seismic Category I SSCs. The staff reviewed the critical damping values specified for seismic analysis of Category I SSCs, and noted that the applicant made no changes to the critical damping values in AP1000 DCD Section 3.7.1.3, between Revision 15 and Revision 17. However, the staff has updated the NUREG-0800 Section 3.7.1 guidance on critical damping, to reference Revision 1 of RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." Both documents were issued in March 2007. RG 1.61, Revision 1, now addresses response-compatible structural damping, electrical distribution system damping (e.g., cable trays), and electrical component damping (e.g., cabinets, panels). The staff noted that the applicant's specified damping values were not higher than the RG 1.61, Revision 1, values in these areas.

The staff issued RAI-SRP3.7.1-SEB1-16, requesting the applicant to specify whether it planned to use the RG 1.61, Revision 1, damping values; or to provide the technical basis for concluding that the damping values the applicant is using will provide sufficient conservatism. In a letter dated May 14, 2009, the applicant submitted its response for each area questioned by the staff:

Response-Compatible Structural Damping

The applicant stated that the HRHF ISRS generated from the analysis are used in evaluating the acceptability of safety-related equipment and components that might be susceptible to HRHF seismic excitation. Acceptability of the equipment is demonstrated by performing an HRHF ISRS seismic test run, after seismic testing to the AP1000 CSDRS ISRS.

In order to address the possibility that the HRHF ISRS may have been underestimated, the applicant included an additional seismic test margin of approximately 30 percent in the HRHF seismic screening evaluation of safety-related equipment vulnerable to HRHF excitation. This is

accomplished by using the 3 percent damping HRHF ISRS in place of the 5 percent damping HRHF ISRS as the required response spectra (RRS) for testing. This approach compensates for the increase in structural response that would have been predicted if the HRHF seismic structural analysis had used 4 percent structural damping instead of 7 percent structural damping.

The staff determined that the 30 percent increase in the RRS is sufficient to compensate for the potential under-prediction of structural response, and is acceptable to meet the intent of the guidance in RG 1.61, Revision 1 (i.e., to use response-compatible structural damping when developing ISRS).

Cable Tray Damping

The applicant stated that the AP1000 design for cable tray support configurations uses construction (Unistrut with bolted connections) covered by the Systematic Evaluation Program (SEP) test program (conducted by ANCO Engineers Inc.). Based on observations during the tests, the high damping values within the cable tray system are provided mainly by the movement, sliding, or bouncing of the cables within the tray. The applicant also stated that the limiting condition for design of the AP1000 standard cable tray supports is for full cable tray weight. The damping value being used for the design of this condition is 10 percent, which is consistent with the value listed in AP1000 DCD Table 3.7.1-1 for full cable trays and related supports. The staff noted that seismic design of full cable trays using 10 percent damping is consistent with the guidance in RG 1.61, Revision 1, and is acceptable.

Electrical Cabinet and Panel Damping

The applicant stated that electrical cabinets and panels employed in safety-related applications are an assembly of structures, subassemblies, and individual components. The electrical cabinets and panels are generally constructed of carbon steel framing members, angle support channels, and panels with a combination of bolted and welded connections designed to support subassemblies and components mounted within. The structural damping of cabinets and panels is a function of the materials, design, mass distribution, and method of interconnection (bolted/welded).

The applicant noted that RG 1.61, Revision 0, defines SSE level damping values as 4 percent for welded steel structures and 7 percent for bolted steel structures; and it is reasonable to perform the analysis of combined bolted and welded structures using an average of the structural damping associated with the bolted or welded steel structures as defined in RG 1.61, Revision 0. In Section 3.7, Table 3.7.1-1 of the AP1000 DCD, Revision 17, the applicant specifies 5 percent damping for electrical cabinets and panels.

The applicant further stated that dynamic structural finite element analyses employ models validated through the use of qualification test program results. The response of the finite element method (FEM) is developed and validated against test data and used as the basis for any modifications that are needed. The results of seismic testing are used in the correlation of dynamic in-equipment response, and the modal and structural damping results from the resonant search test data are used to determine the natural frequency of vibration and

associated structural damping used in model correlation process. In most instances, this leads to the use of 4 percent and 5 percent critical damping in the finite element analysis.

The staff concluded that, although the RG 1.61, Revision 1, guidance is 3 percent damping for electrical cabinets and panels at the SSE analysis level, the applicant has provided an acceptable technical basis for use of higher damping values. For FEM analyses, damping values of 4 to 5 percent are validated by test results. For static coefficient analyses, the use of 5 percent damping is acceptable, when used in conjunction with a 1.5 multiplier on the spectral peak. Although the 1.5 multiplier is intended to provide margin when a multidegree of freedom system or component is analyzed by the static coefficient method, in the case of electrical cabinets and panels, the response is single-mode dominant; the 1.5 multiplier on the 5 percent damping spectral peak would compensate for the difference between 3 percent damping and 5 percent damping.

Based on the applicant's responses and the staff's evaluation, the response to RAI-SRP3.7.1-SEB1-16 is considered acceptable. Confirmatory Item CI-SRP3.7.1-SEB1-16 will track the planned changes in a future revision of the AP1000 DCD.

Shield Building Structural Damping

In the AP1000 DCD, Revision 17, the applicant changed the design of the shield building from reinforced concrete (RC) construction (7 percent SSE damping in AP1000 DCD Table 3.7.1-1) to steel and concrete composite (SC) -filled module construction (5 percent SSE damping in AP1000 DCD Table 3.7.1-1). The staff issued RAI-SRP3.7.1-SEB1-19, part (a), requesting the applicant to define the damping value(s) used for the SC module walls, and to describe how this value is assigned in the [ ] and [ ] models.

The staff also noted that the applicant reduced the shield building concrete modulus (Ec) to 80 percent of nominal value, to account for concrete cracking. The 80 percent value is recommended by the Federal Emergency Management Agency (FEMA) when there is minimal load-induced cracking. Since the 80 percent factor is associated with minimal cracking, the staff noted that use of reduced damping may be appropriate, because damping has been recognized as being a function of the structural response level. At low response levels, lower effective viscous damping has been observed; at high response levels, higher effective viscous damping has been observed. In RAI-SRP3.7.1-SEB1-19, part (b), the staff requested that the applicant submit the technical basis for the damping values assumed. This was identified as Open Item OI-SRP3.7.1-SEB1-19 in the SER with open items.

In its response dated August 26, 2010, the applicant stated that 5 percent structural damping was assumed for the SC modules, including the shield building wall, and 7 percent structural damping was assumed for RC structures. The applicant also stated that these damping values were defined in [ ] and [ ] as a material property defined for each element.

To demonstrate that the assumed damping values for SC and RC are appropriate, the applicant relied on the results of a nonlinear time-history analysis using the [ ] finite element code. In this analysis, concrete was allowed to crack in tension. In Figures RAI-SRP3.7.1-SEB1-19-06 through RAI-SRP3.7.1-SEB1-19-09 of the response, the applicant provided plots of maximum principal stress versus time in the SC, and showed that the

predicted stresses either were close to, or reached, the tensile cracking limit of 43 kips per square foot (ksf) during the progress of the analyzed SSE event. The applicant stated that the use of 5 percent damping was justified if element stresses approached this limit. The applicant also provided a contour plot of maximum principal stresses in the shield building, in Figure RAI-SRP3.7.1-SEB1-19-14 of the response. The applicant stated that the results, at 11.33 seconds, indicate cracking in most of the west side of the shield building wall. Similar contour plots for the RC auxiliary building were provided in Figures RAI-SRP3.7.1-SEB1-19-15 through RAI-SRP3.7.1-SEB1-19-17 of the response, at 7.22 seconds, 8.34 seconds, and 10.28 seconds, respectively. The staff's review of these figures identified that stresses reach the RC tensile cracking limit (36 ksf) in large expanses of the auxiliary building during the SSE event. Based on the applicant's calculations, indicating tensile cracking of concrete for significant portions of the AP1000 NI, the staff finds the applicant's use of SSE-level damping values of 5 percent for the shield building SC wall and 7 percent for RC to be acceptable. Therefore, Open Item OI-SRP3.7.1-SEB1-19 is resolved.

In its August 26, 2010 response, the applicant also addressed the use of concrete stiffness reduction in linear analysis, to account for the effect of concrete cracking. To demonstrate that using a reduced concrete modulus of 0.8 x Ec in the design-basis seismic analysis of the NI is appropriate to account for stiffness reduction due to concrete cracking, the applicant performed nonlinear [ ] analysis, using a smeared concrete cracking model, and compared the results to the results of a linear [ ] analysis, which assumed 0.8 x Ec for the concrete modulus. The applicant submitted additional details of this comparison in its response to related Open Item OI-SRP3.8.3-SEB1-03.

The applicant compared the [ ] results (linear and nonlinear) to linear [ ] NI20 results, in order to validate that the [ ] models are dynamically similar to the [ ] design-basis model. The applicant presented response spectra comparisons, in three orthogonal directions, at the shield building roof in Figures RAI-SRP3.7.1-SEB1-19-11 through RAI-SRP3.7.1-SEB1-19-13 of the response. The comparisons show that the nonlinear [ ] model results are very similar to and are enveloped by the linear model results, which assume 0.8 x Ec. The applicant also provided a plot of stress-strain for a highly stressed element in the shield building (West wall location), in Figure RAI-SRP3.7.1-SEB1-19-02 of the response. The applicant stated that while principal stress values are at or near the assumed cracking threshold (43 ksf), the concrete strains are relatively small; and further stated that the associated secant stiffness would be close to 0.8 x Ec, as shown in Figure RAI-SRP3.7.1-19-01 of the response.

The staff reviewed the applicant's analysis results presented in the response to this open item and in the response to OI-SRP3.8.3-SEB1-03, and determined that the applicant has provided a sufficient technical basis for using a reduced concrete modulus of 0.8 x Ec, to account for stiffness reduction due to cracking. The response is acceptable on the basis that the applicant's comparison of linear (0.8 x Ec) and nonlinear (concrete cracking model) analysis results showed a very good correlation, with the linear model being conservative.

3.7.1.3 Supporting Media for Seismic Category I Structures

In AP1000 DCD, Appendix 3G and accompanying TR-03, the applicant described the supporting media, which define the characteristics of the material providing support for the

AP1000 NI. The AP1000 DCD, Revision 15 was certified for supporting media consisting of HR. In the AP1000 DCD, Revisions 16 and 17, the applicant included a range of FR to SS profiles. For each rock/soil profile, the applicant performed SSI analysis in order to demonstrate the seismic adequacy of the AP1000 plant for the range of soil and rock sites. For the design of seismic Category I structures, a set of six design soil profiles of various Vs was established from parametric studies, as described in AP1000 DCD Appendix 3G and TR-03. The applicant stated that these six profiles are sufficient to envelop sites where the Vs of the supporting medium at the foundation level exceed 304.8 m/s (1000 fps). The design soil profiles include an HR site, an FR site, an SR site, an upper bound soft-to-medium (UBSM) soil site, a soft-to-medium (SM) soil site, and an SS site. The Vs profiles and related governing parameters of the six sites are:

- Hard-rock site - an upper bound case for rock sites using a Vs of 2438.4 m/s (8000 fps).
- Firm-rock site - a Vs of 1066 m/s (3500 fps) to a depth of 36.7 m (120 ft) and base rock at the depth of 36.7 m (120 ft).
- Soft-rock site - a Vs of 731.5 m/s (2400 fps) at the ground surface, increasing linearly to 975.4 m/s (3200 fps) at a depth of 73.12 m (240 ft), and base rock at the depth of 36.7 m (120 ft).
- Upper bound soft-to-medium soil site - a Vs of 430.9 m/s (1414 fps) at ground surface, increasing parabolically to 1034.45 m/s (3394 fps) at 73.2 m (240 ft), base rock at the depth of 36.7 m (120 ft), and ground water at grade level. The initial soil shear modulus profile is twice that of the SM soil site.
- Soft-to-medium soil site - a Vs of 304.8 m/s (1000 fps) at ground surface, increasing parabolically to 731.5 m/s (2400 fps) at 73.15 m (240 ft), base rock at the depth of 36.7 m (120 ft), and ground water is assumed at grade level.
- Soft-soil site - a Vs of 304.8 m/s (1000 fps) at ground surface, increasing linearly to 365.8 m/s (1200 fps) at 73.2 m (240 ft), base rock at the depth of 36.7 m (120 ft), and ground water is assumed at grade level.

The staff reviewed the range of soil profiles and properties identified in AP1000 DCD Revision 17, Section 3.7.1.4, and the iterated Vs profiles presented in Table 3.7.1-4 and Figure 3.7.1-17. In TR-03, Section 4.4, the applicant stated that the range of soil profiles and properties are based on a survey of 22 commercial nuclear power plant sites in the United States. The applicant's survey included sites with Vs ranging from 304.8 m/s (1,000 fps) (SS) to 2438.4 m/s (8,000 fps) (HR). Based on its review, the staff concluded that the applicant has selected a suitable range of site profiles for extending the AP1000 seismic design basis.

3.7.1.4 Conclusion

The NRC staff concludes that the revision to the AP1000 DCD continues to support the seismic design parameters, seismic system analysis, and seismic subsystem analysis for Category I SSCs to meet NRC regulations applicable to the AP1000 DC. The application to amend the AP1000 certified design provides sufficient information to satisfy the applicable requirements of

10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records"; 10 CFR Part 50, Appendix S; and 10 CFR Part 100, "Reactor site criteria," Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," for the seismic design and analysis aspects for Category 1 SSCs to be used in the AP1000 reactor.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each applicant would have to address these issues individually.

3.7.2 Seismic System Analysis

NUREG-0800 Section 3.7.2, "Seismic System Analysis," provides guidelines for the staff to use in reviewing issues related to seismic system analysis. The AP1000 DCD, Revisions 16 and 17 introduced the following significant changes related to AP1000 DCD Section 3.7.2: (1) the applicant performed SSI analysis using the [ ] computer code to extend the AP1000 certified seismic HR design basis to include a range of soil and rock sites; (2) the applicant used 3D shell models instead of 3D stick models for performing dynamic analysis of the NI; (3) the applicant evaluated the effects of HRHF ground motion on the design of AP1000 SSCs; and (4) the applicant used a seismic wave incoherency model in the HRHF analysis, to reduce the effective demand.

The applicant's technical discussion of these changes is incorporated in several sections of the AP1000 DCD and the applicable TRs. The applicant added AP1000 DCD Appendix 3G to document the extension of the seismic design basis to a wide range of soil and rock sites. AP1000 DCD Appendix 3G summarizes the content of TR-03. The applicant also added AP1000 DCD Appendix 3I to briefly summarize the HRHF analysis documented in TR-115. The staff's evaluations of TR-03 and TR-115 are included in Section 3.7.2 of this SER.

The applicant also moved most of the analysis details previously in AP1000 DCD, Revision 15, Section 3.7.2, to the new AP1000 DCD Appendix 3G. The building stick models used in the original HR DC analyses, described in the AP1000 DCD, Revision 15, have been replaced by 3D shell FEMs for the SSI analyses (using [ ]) and for the updated fixed-base analyses (using [ ]). In addition, the equivalent static acceleration methodology, described in the AP1000 DCD, Revision 15 for the detailed design of the buildings, has been replaced by response spectrum analysis (RSA) for the auxiliary/shield building (ASB) and for the CISs.

The applicant's use of a seismic wave incoherency model to effectively reduce HRHF ground motion represents the first application of the ISG-1 on this subject. As a result, the staff performed an independent confirmatory analysis using the applicant's NI20 [ ] model and NI10/NI20 [ ] models. The purpose of the staff's confirmatory analysis was to: (1) evaluate the adequacy of NI20 model for seismic analysis of soil sites and the representative HRHF site; (2) verify the correct implementation of an incoherency model; (3) assess the adequacy of the structures sample set selected by the applicant for HF analysis; and (4) assess overall compliance with ISG-1. The results of the staff's confirmatory analysis effort are described in Section 3.7.2.3.4.2 of this SER.

3.7.2.1 Seismic Analysis Methods

In AP1000 DCD, Revision 17, Section 3.7.2.1, the applicant describes the methods used for performing seismic analyses. The applicant stated that the seismic analyses of the NI are performed in conformance with the criteria in NUREG-0800 Section 3.7.2. RSA, the equivalent static acceleration method, the mode superposition time-history method, and the complex frequency response analysis method are performed for the SSE to determine the seismic force distribution for use in the design of the NI structures, and to develop in-structure seismic responses (accelerations, displacements, and floor response spectra [FRS]) for use in the analysis and design of seismic subsystems. In TR-03, Table 4.2.4-1, the applicant provided a summary of the models and analysis methods used by the applicant in the seismic analyses.

The staff reviewed AP1000 DCD Section 3.7.2.1, and related information in Appendices 3G and 3I, and determined that the applicant's seismic analysis methods are not completely consistent with the latest staff guidance in NUREG-0800 Section 3.7.2, Revision 3 (March 2007). This is discussed in detail in Section 3.7.2.7 of this SER.

The applicant accounted for the effects of SSI by using the [ ] analysis code and used 3D models that accounted for the effects of torsional, rocking and translational responses. The staff finds the [ ] analysis code acceptable for performing SSI analysis because it has been independently benchmarked to standard problems for this type of analysis.

As part of the review of the applicant's SSI analysis methods, the staff performed independent confirmatory analysis using FEMs provided by the applicant. As a result of this effort, the staff identified several modeling errors made by the applicant. The staff's confirmatory analysis is described in Section 3.7.2.4.2 of this SER.

3.7.2.2 Natural Frequencies and Responses

In AP1000 DCD, Revision 17, Section 3.7.2.2, the applicant stated that modal analyses are performed for the shell and lumped-mass stick models of the seismic Category I structures on the NI, as described in Appendix 3G.

The staff reviewed the applicant's seismic analyses models described in AP1000 DCD Section 3.7.2.2, Appendix 3G, and TR-03. The staff issued RAI-TR03-32 and RAI-SRP3.7.1-SEB1-06, requesting the applicant to demonstrate the capability of the NI20 and NI10 models to accurately predict all natural frequencies up to the 33 Hz for the AP1000 CSDRS and up to 50 Hz for the HRHF evaluation spectra. The staff's evaluation for these RAIs is in Section 3.7.2.4 of this SER.

3.7.2.3 Procedures Used for Analytical Modeling

The staff reviewed AP1000 DCD, Revision 17, Section 3.7.2.3, and related information in Appendix 3G. The staff also reviewed TR-03, which provides the detailed information supporting Appendix 3G. In AP1000 DCD Section 3.7.2.3, the applicant indicated that 3D finite element shell models were developed for the coupled shield and auxiliary buildings, and for the CIS. An axisymmetric finite element shell model of the steel containment vessel (SCV) was also developed. These models provide the basis for the development of the dynamic model of

the NI structures. In the dynamic model, the SCV is represented by a lumped mass stick model with properties developed from the SCV axisymmetric model. A separate detailed 3D finite element model of the shield building roof was also developed for detailed design.

The applicant stated that the models of the coupled shield and auxiliary buildings and the CIS are based on the gross concrete section, with the modulus of elasticity reduced to 0.8 times the nominal value, to consider the effect of cracking.

The applicant further stated that seismic subsystems coupled to the overall dynamic model of the NI include the reactor coolant loop model coupled to the CIS model, and the polar crane model coupled to the SCV model. The criteria used for decoupling seismic subsystems from the NI model are taken from Section II.3.b of NUREG-0800 Section 3.7.2, Revision 2.

In TR-03, Section 1.0, the applicant identified the information included in TR-03, in order to update the seismic design basis for AP1000: (1) description of the new 3D shell finite element [ ] and [ ] models; (2) minor structural changes that are significant; (3) the seismic analysis results for a specified range of soil sites; (4) revised envelope ISRS at six reference locations; and (5) the effect of extending the seismic design basis on the seismic design of the NI structures. The staff noted that the only structural change described in TR-03 was the pressurizer compartment redesign. Therefore, in RAI-TR03-001, the staff requested the applicant to describe the other "minor structural changes that are significant" and explain why the changes to the AP1000 design are necessary.

In its response dated January 18, 2007, the applicant stated that the seismic analysis models, NI10 and NI20, have been revised from those reviewed during the HR DC for two types of changes. There are design changes to the AP1000 that include the shorter pressurizer, an increase in spent fuel storage within the existing pit and a revision to the bracing of the shield slab below the discharge stack. There are also changes to the FEM to better reflect the structural configuration. The changes that have been incorporated into the dynamic models, in addition to the redesign of the pressurizer compartment, are:

<u>Design changes</u>

- A design change was made in the spent fuel pool area to permit heavier fuel racks. Masses reflecting the racks and spent fuel were updated. In addition, the water in the fuel pits was modeled as lumped masses instead of solid elements.

- The shield building roof slab bracing was modified from tie rods to cross bracing to improve the seismic response.

<u>Model improvements</u>

- The dish model was modified to incorporate changes in the annulus configuration included in existing AP1000 DCD figures. The annulus tunnel on the west side was deleted and replaced by concrete. In addition, nodes and elements were modified in the lower shield building and upper CIS basemat to be compatible with the revised dish model.

- The core makeup tanks (CMTs) were added as stick models.
- Floors in the CIS model were refined to provide better member force results for use in design.
- Polar Crane Model - Changes made to the model weight (3 percent reduction), updated SCV local stiffness, and inclusion of polar crane truck stiffness.

The applicant stated that these changes were considered minor since the NI building basic configuration was not modified. They reflected structural and model changes that were made during design development.

The staff considered RAI-TR03-001 to be resolved, based on the additional description of changes that the applicant added in Revision 1 and Revision 2 of TR-03. However, the applicant subsequently proposed major design changes to the shield building cylindrical wall, air inlets, and roof in "Design Report for the AP1000 Enhanced Shield Building," March 22, 2010. The staff reviewed the most recent revision of TR-03 (Revision 4, March 2010), and noted that the modeling assumptions used in the dynamic models to simulate the new SC cylindrical wall design are not described. Since this is critical information that is not documented in any of the applicant's formal submittals, the staff requested that the details be added to the next revision of TR-03. This was identified as Open Item OI-TR03-01 in the SER with open items.

In its revised response dated August 26, 2010, the applicant stated that the shield building SC modules are modeled by 3D shell elements using modified stiffness and thickness values to simulate equivalent response in the structure. Equations from AP1000 DCD Section 3.8.3.4.1 were provided in response Figure RAI-TR03-001-01, to describe the procedure for calculating equivalent shell element stiffness and thickness values. In its response, in Figure RAI-TR03-001-02, the applicant provided specific values used in the equations. The staff reviewed the equations used and the numerical results obtained, and concluded that the applicant had properly simulated the stiffness of the SC wall in the [ ] NI10, [ ] NI20, [ ] NI05, and [ ] NI20 models. The applicant also provided a proposed revision to TR-03 to incorporate this information. RAI-TR03-01 and the associated open item are technically resolved. Confirmatory Item CI-TR03-01 will verify the information in TR-03.

In TR-03, Section 4.0, the applicant discussed the dynamic modeling of seismic Category I structures constituting the AP1000 NI. The staff reviewed the applicant's modeling assumptions with respect to concrete material characterization. For the NI, the applicant stated that the concrete modulus of elasticity was reduced to 80 percent of its nominal value, in order to reduce stiffness to simulate cracking. The staff's review of this section found insufficient technical basis for the 20 percent reduction of the modulus of elasticity. In RAI-TR03-05, the staff requested the applicant clarify whether this reduced stiffness was used in the dynamic seismic response analyses for generation of FRS, and in the equivalent static acceleration analyses for design of the structural members. If different stiffness assumptions were used, the staff asked the applicant to provide the technical basis. The staff also requested the applicant to provide the technical basis for using 80 percent, by comparing this to guidance in industry documents such as American Society of Civil Engineers (ASCE) 4-98, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," and to describe any sensitivity studies conducted to

determine the effect of varying the concrete stiffness on ISRS and design of structural members.

In its response dated January 18, 2007, the applicant stated that the reduction to 80 percent is to account for the effects of cracking, as recommended in Table 6-5 of FEMA 356 (Reference: FEMA 356, "Pre-standard and Commentary for the Seismic Rehabilitation of Buildings," (FEMA, November 2000) and that the staff had accepted this basis as part of the AP1000 DCD, Revision 15 review.

The staff evaluated the response and confirmed that during the DC review of the AP1000 founded on HR, the staff had accepted FEMA's recommendation regarding the application of a structural stiffness factor of 0.8 for the seismic analysis of the NI structures.

During the April 2007 audit, the staff requested that the applicant revise its response to clarify that the 0.8 factor for concrete stiffness correlates with test results for essentially uncracked concrete, and does not account for observed or predicted significant cracking (for which a 0.5 factor is more appropriate).

In its revised response dated June 15, 2007, the applicant added that the reduction to 80 percent reflects the observed behavior of concrete when stresses do not result in significant cracking. The applicant also proposed a revision to TR-03, Section 4.0, indicating that concrete structures are modeled with linear elastic uncracked properties, but the modulus of elasticity is reduced to 80 percent of its value to reduce stiffness, to reflect the observed behavior of concrete when stresses do not result in significant cracking, as recommended in Table 6-5 of FEMA 356.

The staff evaluated the response and accepted the applicant's clarification that the use of 0.8 stiffness factor applies when stresses do not result in significant cracking. The staff confirmed that the changes were properly documented in TR-03, Revision 1.

Subsequent to the resolution of RAI-TR03-05, the applicant made major design changes to the cylindrical wall, air inlets, and roof of the shield building. The staff's separate review of the shield building redesign raised questions about the acceptability of the 0.8 factor, since preliminary results presented by the applicant indicate that significant concrete cracking occurs in some areas under seismic loading. The staff requested that the applicant study the sensitivity of the shield building seismic response to a 0.5 stiffness reduction, which is more appropriate when there is significant concrete cracking. The staff had concern that significant concrete cracking could shift the fixed-based frequencies of the shield building, potentially leading to an increase in the seismic demand on the shield building structure and on any systems and components attached to the shield building structure. In its review of TR-03, Revision 4 (March 2010), the staff noted that the 0.8 factor was used for the shield building reanalysis without any discussion or technical justification. This issue is identified as Open Item OI-TR03-05.

In its response dated August 3, 2010, the applicant stated that OI-TR03-05 is addressed in the response to OI-SRP3.7.1-SEB1-19. The staff reviewed the applicant's response to OI-SRP3.7.1-SEB1-19, dated August 26, 2010, and confirmed that it addresses the use of a 0.8 factor for concrete modulus in the design-basis linear seismic analyses. To demonstrate

that using a reduced concrete modulus of 0.8 x Ec in the design-basis seismic analysis of the NI is appropriate to account for stiffness reduction due to concrete cracking, the applicant performed nonlinear [ ] analysis, using a smeared concrete cracking model, and compared the results to the results of a linear [ ] analysis, which assumed 0.8 x Ec for the concrete modulus. The applicant submitted additional details of this comparison in its response to related OI-SRP3.8.3-SEB1-03.

The applicant compared the [ ] results (linear and nonlinear) to linear [ ] NI20 results, in order to validate that the [ ] models are dynamically similar to the [ ] design-basis model. The applicant presented response spectra comparisons, in three orthogonal directions, at the shield building roof in Figures RAI-SRP3.7.1-SEB1-19-11 through RAI-SRP3.7.1-SEB1-19-13 of the response. The comparisons show that the nonlinear [ ] model results are very similar to and enveloped by the linear model results, which assume 0.8 x Ec. The applicant also provided a plot of stress-strain for a highly stressed element in the shield building (West wall location), in Figure RAI-SRP3.7.1-SEB1-19-02 of the response. The applicant stated that while principal stress values are at or near the assumed cracking threshold (43 ksf), the concrete strains are relatively small; and further stated that the associated secant stiffness would be close to 0.8 x Ec, as shown in Figure RAI-SRP3.7.1-19-01 of the response.

The staff reviewed the applicant's analysis results presented in the response to this RAI and in the response to OI-SRP3.8.3-SEB1-03, and determined that the applicant has provided a sufficient technical basis for using a reduced concrete modulus of 0.8 x Ec to account for stiffness reduction due to cracking. The response is acceptable on the basis that the applicant's comparison of linear (0.8 x Ec) and nonlinear (concrete cracking model) analysis results showed a very good correlation, with the linear model being conservative. Therefore, RAI-SRP3.7.1-SEB1-19 is resolved. On the basis that OI-SRP3.7.1-SEB1-19 is resolved, OI-TR03-005 is also resolved.

In TR-03, Section 4.1, the applicant described the modeling assumptions used in the seismic analysis for the water inside the passive containment cooling water storage tank (PCCWST) on the shield building roof. The applicant indicated that a significant percentage of the water mass responds at very low frequency (sloshing), and does not affect the overall building seismic response. Consequently, the applicant concluded that the sloshing water mass could be excluded in the two horizontal directions.

The staff's review of this section found that there was insufficient basis for accepting the applicant's exclusion of sloshing water mass in the dynamic analysis models. In RAI-TR03-007, the staff requested the applicant to provide a detailed technical basis for excluding the low-frequency, water-sloshing mass and to quantify the percentage of water mass in the PCCWST that was excluded.

In a letter dated January 29, 2007, the applicant stated that sloshing of the water in the AP1000 PCCWST was analyzed using a formula for toroidal tanks (Reference J.S. Meserole, A. Fortini, "Slosh Dynamics in a Toroidal Tank," Journal Spacecraft, Volume 24, Number 6, November-December 1987). The fundamental sloshing frequency given by the formula is 0.136 Hz with a modal mass equal to 65 percent of the water mass.

The applicant further stated that AP600 analyses by formula gave frequencies and effective masses similar to those in the AP1000 analyses, and the sloshing formula was confirmed for the AP600 by analyses of a 3D FEM of the water in a rigid tank. For the AP600 design models of the ASB, the applicant found that:

- 60 percent of the water mass was in a sloshing mode. This was included in the AP600 stick model at the elevation of the tank with two masses each with 2 horizontal degrees of freedom.

- The total sloshing mass is 2.6 percent of the mass of the ASB. The stick model results show a maximum absolute acceleration of the sloshing masses of 0.13g, at a frequency of 0.136 Hz.

- The fundamental frequency of the ASB is between 2 and 3 Hz, and the acceleration is 1.1g at the base of the tank.

As a result of the above, the applicant concluded that the low-frequency sloshing mode is not significant to the response of the NI away from the shield building roof and that this conclusion could be extended to the AP1000 design. The horizontal mass participating in the sloshing mode was excluded from the AP1000 3D shell dynamic model of the shield building. However, the applicant considered sloshing in the hydrodynamic loads for the tank wall design.

The staff reviewed the applicant's response and discussed it with the applicant during the April 2007 audit. The applicant stated that the effect of the low-frequency sloshing mode was confirmed to be negligible by performing an analysis of the AP1000 NI stick model without the low-frequency mass, and comparing these results to the results obtained with the low-frequency masses included, provided in Revision 15 of the AP1000 DCD. Comparisons of maximum absolute accelerations, member forces, and FRS indicated there were no significant changes in any of the responses. The staff reviewed the tank sloshing reference and the applicant's calculation. The staff questioned why the percentage of sloshing mass does not go down for the AP1000 versus the AP600, since the increased volume is achieved primarily by making the tank deeper. The applicant agreed to check its estimate of sloshing mass, and provide its conclusions in a supplemental response.

In its revised response dated July 5, 2007, the applicant provided the key dimensions, frequencies and effective masses of the AP600 and AP1000 tanks as shown below.

Parameter	AP600	AP1000	Units
Inside radius of tank	17.5	17.5	feet
Outside radius of tank	38.0	42.5	feet
Average water depth	20.85	22.7	feet
Sloshing frequency	0.139	0.136	Hertz
Ratio of sloshing to total mass	0.66	0.65	

The staff evaluated the response, and concluded that the explanation provided by the applicant to address why the sloshing mass ratio remained unchanged between AP600 and AP1000 was acceptable.

The applicant subsequently made design changes to the PCCWST on top of the shield building. The staff noted that the applicant needed to recalculate the sloshing frequency and sloshing mass to account for any changes in the tank geometry, water depth, and/or free board above the water surface. The staff had concern that overestimating the water sloshing mass could result in an under-prediction of seismic demand for the tank structure. This issue is identified as Open Item OI-TR03-07.

In a letter dated July 12, 2010, the applicant submitted a supplement to its previous RAI-TR03-07 response, stating that the dimensions of the PCCWST were not changed in the enhanced shield building design. The only change affecting the PCCWST is a reduction in elevation by about 1.52 m (5 ft). The applicant also conducted an updated fluid sloshing analysis of the PCCWST, using an [ ] model of the fluid in a rigid tank. The results of the [ ] analysis support the 60 percent assumption for low frequency sloshing modes, as shown below.

Parameter	AP1000		Units
Water weight in 180 degree model	3,337		Kips
Frequency	0.119	0.321	Hertz
Participating weight	1,599	358	Kips
Ratio of sloshing to total mass	47.93	10.73	%

The staff evaluated the applicant's updated analysis results, and concluded that the PCCWST response has a very significant water sloshing component, which has a negligible effect on the overall seismic response of the ASB. On this basis, OI-TR03-07 is resolved.

3.7.2.4 Soil-Structure Interaction

The staff performed a review of the applicant's SSI analyses described in AP1000 DCD Section 3.7.2.4, AP1000 DCD Appendix 3G, and TR-03, using the guidance in NUREG-0800 Section 3.7.2. The design-basis SSI analyses use the AP1000 CSDRS as the seismic input motion; the acceptability of these analyses is evaluated in Section 3.7.2.4.1 of this SER. The staff also performed a review of the applicant's evaluation of HRHF ground motion effects described in AP1000 DCD Appendix 3I and TR-115. Since the staff addressed special considerations for seismic evaluation of HRHF sites in NUREG-0800 Section 3.7.2, under acceptance criteria for SSI, the staff has included the HRHF evaluation in Section 3.7.2.4.2 of this SER.

3.7.2.4.1 Nuclear Island Seismic Analyses using CSDRS Input Motion

In AP1000 DCD Section 3.7.2.4, the applicant stated that the SSI analyses for the FR and soil sites are described in AP1000 DCD Appendix 3G. In AP1000 DCD Sections 3G.4.1

and 3G.4.2, the applicant described the 3D SSI and fixed based analyses. Additional details of these analyses are described in TR-03.

The applicant performed SSI analyses using the computer program [ ]-2000 and the NI20 3D finite element shell model. The SSI analyses were performed for the five soil conditions described in AP1000 DCD Section 3G.3, and reviewed in Section 3.7.1.3 of this SER. The [ ] model included a surrounding layer of excavated soil, as shown in AP1000 DCD Figures 3G.4-3 and 3G.4-4. The seismic input consisted of three statistically independent acceleration time histories (north-south, east-west, and vertical directions), each applied separately. The three resulting time history responses (one for each direction) are combined algebraically at each instant in time. AP1000 DCD Figures 3G.4-5X through 3G.4-10Z provide comparisons of ISRS for the soil cases analyzed. The applicant also performed fixed-base analysis using the [ ] model, to simulate HR conditions (i.e., Vs greater than 8,000 fps).

Selection of Soil Cases

The staff reviewed the applicant's description of site studies and selection of soil cases described in Section 4.4.1.2 of TR-03. The staff's review of Tables 4.4.1-1A and 4.4.1-1B of TR-03 identified that the applicant used three soil/rock degradation models in its parametric studies for selecting site conditions: Seed and Idriss 1970 soil/rock degradation curves; Idriss 1990 soil degradation curves; and EPRI 1993 soil degradation curves. In RAI-TR03-10, the staff requested the applicant to provide the technical basis for using these different soil degradation models for its parametric studies.

In its response dated January 18, 2007, the applicant stated that SSI analyses on rock sites for both the AP600 and the AP1000 use the rock degradation curve recommended by Seed and Idriss (Reference: Seed, H.B. and I.M. Idriss, "Soil Moduli and Damping Factors for Dynamic Response Analysis," Report Number. EERC [Energy and Environmental Research Center] 70-14, Earthquake Engineering Center, University of California, Berkeley, CA, 1970). This was applied in SSI analyses for the HR, FR and SR sites. The applicant further stated that SSI analyses on soil sites for the AP1000 used the latest soil degradation curve recommended by EPRI (Reference EPRI TR-102293, "Guidelines for Determining Design Basis Ground Motions," 1993). This was applied in SSI analyses for the UBSM, SM, and SS sites. Two sets of degradation curves were used in the AP600 studies. The early analyses used the degradation curve recommended by Seed and Idriss. Later analyses performed to address NRC questions used the later soil degradation curve recommended by Idriss (Reference Idriss, I.M., "Response of Soft Soil Sites during Earthquakes," H. Bolton Seed Memorial Symposium Proceedings, May 1990). The applicant provided a proposed revision to AP1000 DCD Section 3.7.1.4 and additional figures for inclusion in the AP1000 DCD.

The staff evaluated the response and noted a number of issues in need of further clarification:

1. The EPRI 1993 model shown in the proposed AP1000 DCD Figure 3.7.1-16 indicates hysteretic damping levels greater than 15 percent. In NUREG-0800 Section 3.7.2.4, the staff imposed a limit of 15 percent on hysteretic damping. The applicant should provide the final iterated Vs profile and damping levels reached throughout the soil column, for

each case analyzed for site response, and show that damping levels do not exceed the 15 percent limit.

2. The EPRI 1993 model is generally considered appropriate for cohesionless soils. The model is not considered appropriate for cohesive fine-grained soils. The AP1000 DCD should indicate the criteria to be used by the COL applicant to evaluate the appropriateness of this degradation model for site-specific application.

3. The AP1000 DCD should include the strain-iterated Vs profiles that need to be compared to the site-specific velocity profiles generated by the COL applicant.

During the April 2007 audit, the applicant agreed to supplement its response by identifying the bounds of the strain-iterated Vs profiles. The applicant also agreed to describe how a COL applicant confirms that its site is enveloped by the generic seismic design basis. In its revised response dated July 5, 2007, the applicant stated that: (1) the soil profiles used in the generic analyses will be added to AP1000 DCD Section 3.7.1.4; (2) additional clarification of how to confirm that a specific site is enveloped by the generic seismic design basis will be provided in proposed revisions to AP1000 DCD Section 2.5.2; and (3) TR-03, Section 4.4.1.2, will be revised to include the description and table of degraded properties for each soil profile.

During the May 2008 audit, the staff and the applicant agreed that the site-specific Vs profile should be based on low-strain minimum measured values; and that a criterion is needed to define the acceptable variation in Vs when the site-specific soil profile shows an inversion (i.e., soft material under hard material). These issues are addressed under RAI-SRP2.5-RGS1-15.

During the April 2009 audit, the staff requested the applicant to provide clarification in the AP1000 DCD concerning limitations on the use of two dimensional (2D) [ ] analyses to address site-specific deviations from the certified design site parameter envelope. In a letter dated May 15, 2009, the applicant submitted a proposed revision to AP1000 DCD Section 2.5.2.3 to provide this clarification:

> The Combined License applicant may identify site-specific features and parameters that are not clearly within the guidance provided in subsection 2.5.2.1. These features and parameters may be demonstrated to be acceptable by performing site-specific seismic analyses. If the site-specific spectra at foundation level at a hard rock site or at grade for other sites exceed the certified seismic design response spectra in Figures 3.7.1-1 and 3.7.1-2 at any frequency (or Figures 3I.1-1 and 3I.1-2 for a hard rock site), or if soil conditions are outside the range evaluated for AP1000 design certification, a site-specific evaluation can be performed. These analyses may be either 2-D or 3-D.
>
> 3-D [ ] analyses will be used to quantify the effects of exceedances of site-specific GMRS compared to the CSDRS, or the HRHF GMRS at a hard rock site (DCD Figures 3I.1-1 and 3I.1-2), or in cases where the site specific velocity soil profiles do not fall within the range evaluated for the standard design. 2-D

analyses are performed for parameter studies. Results will be compared to the corresponding 2-D or 3-D generic analyses.

The staff reviewed the applicant's proposed revision to AP1000 DCD Section 2.5.2.3, and the applicant's response to RAI-SRP2.5-RGS1-15, and concluded that the open technical issues had been adequately addressed. The applicant clarified the limitations on the use of 2D [ ] analyses to address site-specific deviations from the certified design site parameter envelope; and also provided additional criteria that must be satisfied at a specific site in order to be covered by the AP1000 generic soil site analyses. Therefore, RAI-TR03-10 is now identified as Confirmatory Item CI-TR03-10, which will be resolved upon confirmation of information in a future revision to the AP1000 DCD. In Section 4.4.1 of TR-03, the applicant stated that many results and conclusions from the AP600 soil studies are applicable for the AP1000. In RAI-TR03-14, the staff requested the applicant to describe which results and conclusions from the AP600 soil studies are applicable to the AP1000.

In a letter dated January 18, 2007, the applicant stated that the AP600 design is based on enveloped results from analyses for four soil conditions (HR, SR, UBSM, and SM). These four soil cases were selected from the parametric analyses summarized in Section 4.4.1 of TR-03. The AP600 soil studies demonstrated that these four cases would bound sites having soil with Vs exceeding 1,000 fps. Parameters selected for the design soil cases from these analyses were:

- Depth to bedrock of 36.7 m (120 ft)
- Water table for the UBSM and SM cases up to grade
- Parabolic variation of Vs with depth for the UBSM and SM cases

The applicant stated that parametric analyses of the AP1000 were performed for six soil cases, as described in TR-03, Section 4.4.1.2. These analyses used the same assumptions for depth-to-bedrock, depth-to-water table, and variation of Vs with depth as used in the AP600 analyses. These analyses confirmed that the response of the AP1000 was similar to that of the AP600 for these soil cases, with the AP1000 fundamental response occurring at lower frequencies due to its increased height 7.6 m (25 ft) and mass (10 percent).

The staff evaluated the RAI response and concluded that the applicant provided a sufficient description of the design parameters derived from the AP600 analyses in TR-03, Section 4.4. On this basis, RAI-TR03-14 was resolved.

In TR-03, Section 4.4.1, the applicant concluded that some effects (water table, soil layering, soil-degradation model, etc.) are not significant for the seismic response of the NI structures. The staff's review of this section found that the applicant did not provide sufficient basis for making the above conclusions. In RAI-TR03-15, the staff requested the applicant to provide the technical basis for drawing these conclusions for the AP1000. In addition, the staff requested that the applicant demonstrate that the combination of these effects is also insignificant for the seismic response of the NI structures.

In a letter dated August 20, 2008, the applicant submitted a comprehensive response to address the staff's questions. The referenced figures and tables were submitted as part of the RAI response. Paraphrasing the applicant's response:

Revised TR-03 Section 4.4.1.1 provides additional technical basis for the selection of the soil parameters used in the AP1000 3D [ ] design cases. The soil cases selected for the AP1000 use the same parameters on depth-to-bedrock, depth-to-water table and variation of Vs with depth as those used in the AP600 design analyses. The parameters used for the AP1000, based on the results and conclusions from the AP600 soil studies, are summarized in Table 4.4.1-1A. The AP600 soil studies considered variations of the parameters and combinations thereof in establishing the design soil profiles. AP1000 has a footprint identical to that of the AP600 and is similar in overall mass. The height of the shield building is increased by about 20'. The total weight of the NI increases by about 10 percent. Parametric analyses of the AP1000 were performed for six soil cases, as described in Section 4.4.1.2. The AP1000 response is very similar to AP600, except that the fundamental response occurs at lower frequencies due to the increased height and mass of the NI. Based on the similar response in these analyses, it is concluded that the governing parameters obtained for the AP600 soil studies are also applicable to the AP1000.

The applicant addressed soil degradation in RAI-TR03-10. Tables of strain-iterated Vs used in the generic analyses are shown in Table 4.4.1-3 of TR-03. Figure RAI-TR03-15-1 shows the bounds of these strain-iterated Vs profiles. The combination of effects of the different soil parameters is reflected in these bounds. Figure RAI-TR03-15-2 shows how a COL applicant could demonstrate that the site is enveloped by generic seismic design basis. The applicant would define its site geotechnical parameters as defined in AP1000 DCD Section 2.5 and would justify why the site is within the bounds of the AP1000 generic analyses that have been considered in this TR. These parameters would include the soil profiles used in the probabilistic seismic hazard analysis (PSHA), which could then be compared to Figure RAI-TR03-15-1. Subsequent discussions between the COL applicant and the NRC may uncover a parameter for which more justification is required, in order to show that the impact of this parameter on the response is small. This justification could be done with the AP1000 2D model. An example of how a 2D parametric study would be used is shown in Figure RAI-TR03-15-3 and RAI-TR03-15-4. If the parametric 2D [ ] studies show that the effect could be significant (e.g., 90 percent of the design spectrum, see Figure RAI-TR03-15-4) when compared to the 2D design spectra, a 3D [ ] study would then be performed. If the 3D [ ] analyses show some exceedances at the critical locations, the applicant would then proceed to show that sufficient margin exists in the design to accommodate these exceedances.

The effect of water table on the seismic response of the NI structures is shown in Figures RAI-TR03-15-5 through RAI-TR03-15-7. Case 1 (SM) shows the results for the SM generic case profile, which assumes water table at grade. Case 2 (SM-NW) results are for the same soil condition except the water table is below the bottom of the soil profile at 36.7 m (120 ft) below grade. As can be seen, there is negligible difference between the two cases for the horizontal response. The vertical response due to the design profile with the water table at grade (Case 1) is more conservative than that for the dry soil profile (Case 2). This result is similar to the results in the AP600 study, which are summarized in TR-03, Section 4.4.1.1. Thus, the generic analyses are conservative for sites with a lower water table.

The staff determined that the information presented in the applicant's revised response to RAI-TR03-15, and supplementary information in the RAI-TR03-10 response related to soil degradation models, are sufficient to address the staff's questions. The staff also confirmed that all proposed revisions to TR-03 have been formally submitted in Revision 4. Therefore, RAI-TR03-15 is resolved.

Seismic Analysis Results

During its review of TR-03, the staff identified that equivalent static analysis was employed to calculate maximum member forces for detailed design of the NI structures, using acceleration versus height profiles obtained from the time history analyses. The staff's separate review of TR-09, "Containment Vessel Design Adjacent to Large Penetrations," identified that the SCV is designed for equivalent static accelerations determined from the fixed-base NI stick model, tabulated in AP1000 DCD Table 3.7.2-6, which are representative of the HR condition. In RAI-TR03-16, the staff requested the applicant to: (1) identify the site condition(s) selected to develop the equivalent static acceleration profile used to perform the equivalent static analysis; and (2) discuss whether the seismic loads used for design of the SCV envelop both the fixed-base HR condition and the worst-case condition from all soil sites considered.

The applicant's initial responses to this RAI did not fully address the staff's concern. As an alternative, the staff requested the applicant to provide a direct comparison of the equivalent static analysis results to time history analysis or RSA results. During the October 2007 audit, the applicant indicated it had switched the detailed evaluations of the CIS and ASB from equivalent static analysis to RSA. However, for the SCV, the applicant did not address whether the equivalent static acceleration method yields conservative results, when compared to RSA or time history analysis.

At the April 2009 audit, the applicant presented a comparison of results for the SCV, between equivalent static analysis and a mode superposition time history analysis, at major containment penetrations. The comparison showed that the equivalent static analysis results are higher than the time history results. The applicant agreed to revise its RAI response, to include the information presented at the audit.

In its revised response dated May 15, 2009, the applicant stated that the equivalent static acceleration analyses of the containment vessel (CV), described in TR-09, use a finite element shell model with a refined mesh in the area adjacent to the large penetrations (Figure 2-6 of TR-09). A reanalysis was performed using the same methodology on the coarse-mesh model of the SCV. The applicant performed a time history analysis of the coarse-mesh model, selecting information for the regions immediately surrounding the large penetrations, as shown in Figure RAI-TR03-016-001, for the purpose of comparing the loads from equivalent static analysis and time history analysis. The effects of the missing mass in the time history analysis were incorporated by an algebraic sum of the stress intensities from a run with the left-out mass accelerated at zero period acceleration (ZPA) and the modal superposition time history analysis. Figures RAI-TR03-016-002 through RAI-TR03-016-005 (attached to the RAI response) compare the stress intensity for individual elements surrounding the major penetrations. The applicant stated that the results from these analyses show that equivalent static analysis consistently produced higher stresses than the time history results. The staff reviewed the analysis

comparisons and concluded that the equivalent static acceleration results for the SCV are conservative, when compared to time history results. Therefore, RAI-TR03-16 is resolved.

During its review of Section 6.2 of TR-03, the staff identified a number of editorial and technical items in need of clarification or explanation. In RAI-TR03-21, parts (b), (c), and (e), the staff requested the applicant to provide technical clarifications. Parts (a) and (d) were editorial.

> (b) TR-03, Section 6.2, states "For those local flexible structures that are amplified, apply an additional acceleration to these structures equal to the difference between the average uniform amplified component accelerations and rigid body component equivalent static accelerations. These accelerations are to be considered in local design of the flexible portion of the structure but do not need to be considered in areas of the structure away from the local flexibility. They can be applied in a series of individual load vectors." It is not obvious to the staff how this methodology has been implemented, and whether the effects of increased accelerations on locally flexible structures can be ignored in areas of the structure away from the locally flexible structures. The sum total of all the flexible masses times the corresponding acceleration increments may impose greater-than-negligible additional loads on the overall structure, in the two horizontal directions and in the vertical direction. Therefore, the applicant is requested to (1) describe in greater detail the implementation of this methodology, including a numerical example; and (2) provide a quantitative technical basis for the conclusion that the effects of increased accelerations on locally flexible structures can be ignored in areas of the structure away from the locally flexible structures."
>
> (c) TR-03, Section 6.2, states "The vertical equivalent static seismic accelerations at (Shield Bldg) elevations 89.9 m (294.93 ft) and 101.5 m (333.13 ft) are obtained directly from the maximum time history results by taking the average of locations at opposite ends of a diameter. The vertical accelerations from the 3D finite element model at the shield building edges at these elevations are significantly influenced by the horizontal loading. If they are used for the vertical equivalent accelerations, the horizontal response would be double counted in the vertical direction." It is not obvious to the staff how this methodology has been implemented, and whether it is even appropriate. Therefore, the applicant is requested to submit a numerical example, based on elevation 101.5 m (333.13 ft) of the SB, to demonstrate the implementation of this methodology. In this example, please also include the vertical acceleration value that would be obtained if this methodology was not implemented."
>
> (e) TR-03, Section 6.2, under the heading "Seismic Accelerations for Evaluation of Building Overturning," states "The dynamic response of the structure affecting overturning and basemat lift off is primarily the first mode response at about 3 Hertz on hard rock. This reduces to about 2.4 Hertz on soil sites as shown in the 2D [ ] and [ ] analyses. The higher auxiliary building accelerations of Table 6.2-2 are not considered in overturning since they are from higher frequency modes greater than 2.4 Hertz. Amplified response of individual walls in the Auxiliary Building and the IRWST [In-Containment Refueling Water Storage Tank] need not be considered since they are local responses that do not effect overturning." For the overturning analysis, the staff is concerned that the methodology employed may not predict an overall moment on the basemat that envelops the maximum overturning moment for all site conditions. The

applicant is requested to provide its technical basis for the conservatism of the methodology employed.

In a letter dated April 5, 2007, the applicant provided its initial response to this RAI. For part (b), the staff required additional clarification concerning how the applicant determined the uniform acceleration values applied to the whole structure and the additional acceleration increments applied to the flexible areas.

For part (c), the applicant stated that a seismic component associated with the rotational response of the PCCWST should also be included, in addition to the translational seismic acceleration component, and that the rotational response of the PCCWST would be addressed in the redesign of the shield building roof.

For part (e), the applicant proposed that it be deferred to the staff's review of APP-GW-GLR-044, Revision 0, "Nuclear Island Basemat and Foundation."

At the October 2007 audit, partly in response to part (b) of this RAI, the applicant presented results from an RSA of the coupled ASB/CIS, using the refined [ ] model. The applicant had decided to use these RSA results as the basis for detailed design of the ASB and CIS. At the time, the applicant stated that switching to RSA resolved parts (b) and (c) of this RAI.

During the May 2008 audit, the staff requested that the applicant demonstrate that the seismic RSA using the fixed base NI05 model is sufficient to capture additional amplification due to rocking. The applicant agreed to compare loads at the top of the shield building, between time history analysis, which includes rocking, and RSA, which does not.

On August 20, 2008, the applicant submitted its revised response to parts (b) and (c) of this RAI. The staff concluded that the questions raised in part (b) of this RAI were no longer applicable. The staff confirmed that TR-03, Revision 2, Section 6.4, clearly identified that RSA is used for the ASB design and the CIS design. Therefore, part (b) was resolved.

For part (c), the applicant presented a comparison of the bending moments in the beams at the top of the shield building, and the forces and moments in the PCS vertical wall, between time history and RSA results. In all cases, the RSA is conservative when compared to the time history analysis, confirming that conservatism in the RSA that will account for rocking. The staff concluded that the comparisons sufficiently demonstrated the conservatism of the RSA results. Therefore, part (c) was resolved.

Part (e) of this RAI, concerning the conservatism of the overall moment on the basemat, is addressed in Section 2.6.1.2 of TR-85, "Nuclear Island Basemat and Foundation," and is tracked under the staff's TR-85 evaluation. This issue is considered resolved with respect to the TR-03 evaluation. Therefore, RAI-TR03-21 was resolved.

The staff reviewed the applicant's seismic displacement results presented in TR-03, Section 6.3. The maximum seismic deflections obtained from the fixed-base time history analysis and the [ ] analyses are given in Tables 6.3-1 to 6.3-3 for the ASB, CIS, and SCV, respectively. The staff determined that a number of clarifications were needed before the staff could complete

its review. In RAI-TR03-22, the staff requested the applicant to: (1) clarify whether the deflections in the tables are a consistent set, based on the worst-case time history result, or are an envelope of maximum deflections from all the time history results; and (2) compare the tabulated deflections to the corresponding deflections obtained from the equivalent static acceleration analyses, and explain any significant differences.

In its response dated January 29, 2007, the applicant stated that the deflections given in Tables 6.3-1 to 6.3-3 are the envelope of maximum relative deflections from all of the time history results for the soil and HR cases. Displacements at different nodes for the soil cases have been obtained relative to the translation of a reference node at the bottom of the foundation and near the center of the basemat. Deflections for the HR case are relative to the fixed base at foundation level.

The applicant further stated that the deflections given in these tables have been revised to remove drift, by adding a small constant acceleration to the response acceleration at every time step for the first 0.05 seconds of the time history. If baseline correction is not performed, a residual drift in displacement time histories will be obtained at the end of the seismic excitation. The applicant provided Tables RAI-TR03-022-1 to RAI-TR03-022-3 in its response, showing the revised relative displacements. The applicant also stated that it is not possible to compare equivalent static displacements to the time history displacements for the soil cases. The time history results include rocking about the base, while the equivalent static analysis has a fixed base.

The staff questioned the approach the applicant had used to eliminate drift and, following discussions of this issue during audits in 2007, and 2008, the applicant submitted a revised RAI response, in a letter dated August 20, 2008. The applicant revised the approach for eliminating drift. The new approach calculates displacements internally within the [ ] program, based on an analytical complex frequency domain approach that uses inverse fast Fourier transforms to compute relative displacement histories, instead of double numerical integration in the time domain for computing absolute displacement time histories from absolute acceleration time histories. The analytical approach is more accurate than a typical baseline correction (time integration) algorithm. The applicant also submitted a proposed revision to TR-03, Section 6.3, "Seismic Displacement Calculation," adding more detail about the analysis methodology and identifying that the ACS [ ] RELDISP module is used for this calculation.

The applicant also indicated in its response that it had switched to seismic RSA and is not using equivalent static analyses; and consequently the staff's initial request for comparison of dynamic results to equivalent static analysis results is no longer applicable. The applicant also submitted a proposed revision to TR-03, Section 6.3, covering this change.

The staff reviewed the response and found the applicant's revised approach to eliminate drift acceptable, because it is mathematically rigorous. For comparison of displacements, the staff noted that RSA is only applied to the ASB and CIS, not to the SCV. Thus, this issue remained unresolved for the SCV. The staff confirmed that TR-03 had been appropriately revised in Revision 3, resolving the drift issue. The applicant also submitted a detailed comparison of time history results to equivalent static acceleration results for the SCV, in a revised response to RAI-TR03-16, demonstrating the conservatism of the equivalent static analysis for the SCV. As

a result, the staff considered the static vs. dynamic issue resolved for the SCV. Therefore, RAI-TR03-22 was resolved.

In a letter dated September 10, 2010, the applicant submitted revised responses to RAI-TR03-22 and related RAI-TR03-37. These responses identified alternate methods that the applicant has used to calculate relative displacements. The applicant identified two methods, in addition to the ACS [ ] RELDISP module, for inclusion in the next revision of the AP1000 DCD and the next revision of TR-03. The proposed AP1000 DCD additions, included in the response to RAI-TR03-37, are as follows:

DCD 3G.4.1 "[ ] Fixed Base Analysis"

> [ ] is used to calculate the maximum relative deflection to the nuclear island for the envelop case that considers all of the soil and hard rock site cases. Synthesized displacement time histories are developed using the envelope seismic response spectra from the six site conditions (hard rock, firm rock, soft rock, upper-bound soft-to-medium, soft-to-medium, and soft soil). Seismic response spectra at nine locations are used (4 edge locations, 1 center location, and 4 corner locations). It is not necessary to adjust for drift since deflections relative to the basemat are calculated, and the drift would be subtracted from the results.

DCD 3G.4.2 "3D [ ] Analyses"

> Westinghouse has adopted the approach that calculates displacements internally within the ACS [ ] program based on an analytical complex frequency domain approach that uses inverse Fast-Fourier Transforms (FFT) to compute relative displacement histories instead of double numerical integration in the time domain that computes absolute displacement time histories from absolute acceleration time histories.
>
> The relative displacement time history is calculated using ACS [ ] RELDISP module. The complex acceleration transfer functions (TF) are computed for reference and all selected output nodes. The relative acceleration transfer function is calculated by subtracting the reference node TF from the output node TF. The relative displacement transfer function is obtained by dividing the circular frequency square (ω^2) for each frequency data point. The relative displacement time history is obtained by taking the inverse FFT.
>
> Relative displacements are calculated between adjacent buildings and the nuclear island using soft springs between the buildings. The spring stiffness is very small so that it does not affect the dynamic response. These calculations are performed using 2-D models and the [ ] 2000 code. The relative deflection is calculated using the maximum compressive spring force and the stiffness value.

The applicant also proposed comparable revisions to TR-03 in the response to RAI-TR03-22. The staff determined that the additional methods used by the applicant to calculate relative

displacements are technically correct, and do not require any correction for drift. Confirmatory Item CI-TR03-22 will be resolved upon confirmation of information in a future revision of TR-03. In addition, Confirmatory Item CI-TR03-37 will be resolved upon confirmation of information in a future revision of the AP1000 DCD.

The staff reviewed the comparison of the NI10 and NI20 seismic analysis models, described in TR-03, Appendix C. The staff's review identified the need for a number of clarifications and explanations of the results presented. In RAI-TR03-32, the staff requested the applicant to provide these clarifications and explanations.

The staff and the applicant discussed the issues raised in this RAI at audits in 2007, 2008, and 2009. The applicant submitted several revisions to its RAI response, to address the staff's original and follow-up questions. Following the April 2009 audit, the only remaining technical issue was whether the NI20 model refinement is sufficient to represent vibration modes up to 33 Hz are potentially excited by the CSDRS ground spectrum input. The staff was concerned that, if the dynamic analysis model(s) of the AP1000 do not accurately predict the amplified response of flexible regions, then the ISRS at those locations may be underestimated. The staff initiated an independent comparison of modal properties between the [ ] NI10 model and the [ ] NI20 model. Based on the preliminary results of the staff's confirmatory analyses, the staff requested the applicant to demonstrate that all walls, floors, and roof slabs with a fundamental plate vibration frequency less than 33 Hz are adequately represented in the NI20 model, such that an [ ] NI20 modal analysis will capture these vibration modes. If this is not the case for specific walls, floors, or roof slabs, the staff requested that the applicant develops an approach to generate the ISRS that consider the additional amplification in the middle of the wall, floor, or roof slab.

In TR-03, Revision 4 (March 2010), Section 4.2.4, the applicant stated that the NI05 model was reviewed to identify flexible regions that may produce amplified response spectra. The applicant concluded that the NI20 model was too coarse in some areas to pick up all local vibration modes up to 33 Hz, based on comparison to NI05 modal analysis results.

Consequently, the seismic response in the middle of some wall, floor, and roof panels is underestimated, leading to nonconservative ISRS for subsystem design. To address this, the applicant proposed a method of evaluating these areas using the more detailed NI05 model to evaluate flexible regions. The staff's review of the proposed method found that that there was insufficient description of the proposed method and that an example case (including results) would be helpful in understanding the implementation. This issue is identified as Open Item OI-TR03-32.

In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-TR03-32. The applicant stated the NI05 model had been reviewed for flexible regions where out-of-plane response may occur at frequencies less than 33 Hz. The applicant noted that each of the regions reviewed have a higher mesh refinement than the NI20 model. The regions, which have flexible areas, are evaluated in one of two ways:

1. Flexible areas that were previously identified (TR-03, Revision 4, Table 4.2.4-10) have amplified response spectra developed from the envelope of the time history analysis results for the HR and soil sites.

2. Flexible regions, which require a detailed analysis to obtain the amplified response spectra use input directly from time history analysis. The NI05 finite element model is used to capture out-of-plane flexibilities that, because of mesh refinement, the NI10 and NI20 models could not capture. The resulting nodes have been designated with (NI05) to distinguish that the amplified response spectra come from that model.

This applicant identified proposed revisions to TR-03, to document the methods and results. The staff reviewed the flexible regions identified in Tables RAI-TR03-032-2, RAI-TR03-032-3, and RAI-TR03-032-4 of the RAI response, and the ISRS comparisons (NI05 amplified versus NI10/NI20) shown in Figures RAI-TR03-032-7 to RAI-TR03-032-13 of the RAI response. Based on its review, the staff finds the applicant's method for identifying flexible regions and modifying the ISRS to be acceptable. By using the mesh refinement of the NI05 model, the applicant was able to locate and evaluate flexible regions of the NI structures that were inadequately modeled in the less refined NI20 and NI10 models. RAI-TR03-32 and the associated open item are technically resolved. Confirmatory Item CI-TR03-32 will track the formal submittal of the TR-03 revision.

The staff reviewed TR-03, Section 4.2.4, which summarizes the applicant's seismic analysis models and methods used for the AP1000 design. In Table 4.2.4-1 of TR-03, the applicant summarized the type of structural models, analysis methods, and computer codes used in the evaluations to extend the NI seismic analyses to soil sites. In the table, the applicant stated that the 2D finite element lumped-mass stick model of the ASB was analyzed using the [ ] Code, by time history analysis method for the purpose of parametric studies to establish the bounding generic soil conditions. However, during its review of the responses to other RAIs, the staff noted that 2D seismic analyses were apparently used for other purposes also. In RAI-TR03-34, the staff requested the applicant clarify the information provided in Table 4.2.4-1, and update this table, as needed, to identify all applications of 2D seismic analysis, and how the results were used.

In its response dated July 5, 2007, the applicant stated that Table 4.2.4-1 had been revised to show the additional seismic models and analyses identified. The revision to the table also added the polar crane models and the CV shell model, included in the response to RAI-TR03-20. During the May 2008 audit, the staff verified that TR-03, Revision 1 included the revised Table 4.2.4-1, documenting the use of 2D analysis models. However, additional errors were found in the table. In a letter dated August 20, 2008, the applicant submitted a proposed revision to TR-03 Table 4.2.4-1. Confirmatory Item CI-TR03-34 will track the formal submittal of the final revision to TR-03 Table 4.2.4-1.

3.7.2.4.2 Nuclear Island Seismic Analysis using HRHF Input Motion

Subsequent to NUREG-1793 for the AP1000 DCD, Revision 15, the applicant added AP1000 DCD Appendix 3I in Revisions 16 and 17, in order to address the adequacy of the AP1000 seismic design for ground response spectra typical of CEUS HR sites, which are "rich in the high frequency range." These sites are referred to as HRHF sites. The applicant's technical basis for AP1000 DCD Appendix 3I is TR-115, "Effect of High Frequency Seismic Content on SSCs."

In May 2008, the staff issued ISG-1 on acceptable methods to demonstrate seismic adequacy for HRHF ground spectra. The four key elements of the guidance are:

- Use of the staff-accepted Abrahamson coherency function, to reduce the effects of the high-frequency ground motion.
- Use of a staff-accepted computer code (e.g., ACS [ ]) specifically developed to include the effects of incoherency.
- Use of building structural models sufficiently refined to adequately predict modal response up to 50 Hz.
- Selection of an adequate sampling of SSCs for detailed evaluation of response to the HRHF ground spectra.

The staff reviewed AP1000 DCD Appendix 3I and TR-115 using the elements of the ISG-1, in full consideration that the applicant's submittal represent the industry's first attempt to implement ISG-1.

The staff reviewed the introduction to TR-115, Revision 0, Section 1.0, and noted that the first paragraph stated that the purpose of the report is two-fold: (1) to confirm that high frequency seismic input is not damaging to equipment and structures qualified by analysis for the AP1000 CSDRS; and (2) to demonstrate that normal design practices result in an AP1000 design that is safer and more conservative than that which would result if designed for the high frequency input. The staff found that the above statements, made by the applicant, were too generic in nature, and required a qualification that they apply only to the HRHF spectra actually used in the analyses. The staff also noted that the last paragraph to the introduction section of TR-115 needed to be similarly qualified. In RAI-SRP3.7.1-SEB1-02, the staff requested that the applicant revise the stated purpose of TR-115, accordingly.

In a letter dated April 25, 2008, the applicant proposed changes to the introduction section of TR-115, to satisfy the staff's concern. The staff evaluated the RAI response and the proposed revisions to TR-115, and found them acceptable. The staff subsequently confirmed that TR-115, Revision 1, included the proposed revisions.

Although the applicant clarified the purpose of TR-115, the staff determined that the report contained insufficient information regarding site parameter requirements. The staff requested that the applicant specifically identify in TR-115 the minimum Vs of the underlying medium that must be satisfied in order to reference the results in TR-115, and also provide the technical basis for this determination. The staff noted that the definition of an HR site in the AP1000 DCD is a site with a minimum Vs of 2438.4 m/s (8,000 fps).

In a letter dated September 12, 2008, the applicant responded that the only requirement that COL applicants must demonstrate, to be covered by TR-115, is that their site GMRS is enveloped by the HRHF spectra. The applicant stated that sites with high Vs have higher loads due to a higher frequency than those with lower Vs, and sites that are enveloped by the HRHF input spectra, but have lower Vs, will have lower HRHF seismic loads than those used in the evaluation reported in TR-115.

The staff evaluated the supplemental response, and determined that the applicant's statement, that only a spectrum comparison is necessary, has no established technical basis. Softer material beneath the foundation will shift spectral peaks; whether the results for softer materials are enveloped by the HR results needs to be demonstrated. Based on the above assessment, the staff submitted Supplement 2 to RAI-SRP3.7.1-SEB1-02, requesting the applicant to address the following:

(a) Describe in detail the modeling of underlying media and any side media in the special [ ] analyses of the HRHF GMRS. How many cases were analyzed? Describe each case and the purpose for each case.

(b) What is the Vs associated with each of the media included in the [ ] analyses?

(c) How was the seismic motion at the surface developed for input to the [ ] analyses? Was the HRHF GMRS applied directly as surface motion, or was the surface motion developed from the HRHF GMRS applied at the NI foundation level? If the latter, describe in detail the method used to calculate the surface motion.

(d) Define numerically the range of Vs of the underlying media for which the special [ ] analyses are valid. Provide a detailed technical basis for this determination (e.g., results from parametric studies, previous documented studies, documented test results, "expert" judgment, etc.).

(e) For all COL applications that reference AP1000 DCD Appendix 3I and/or TR-115, are the site characteristics enveloped by the range of Vs defined in (d) above?

In a letter dated February 19, 2009, the applicant responded to RAI-SRP3.7.1-SEB1-02 (Supplement 2). The applicant presented a table of Vs versus depth for the single HRHF analysis conducted, but also restated its contention that only a spectral comparison is required. The staff found the applicant's response to Supplement 2 to be unacceptable, and discussed this with the applicant in a teleconference on March 5, 2009. The applicant agreed that it is necessary for a specific site to satisfy both the response spectra criteria and also the Vs profile, in order to be covered by the analysis reported in TR-115.

In a letter dated April 14, 2009, the applicant revised its response to RAI-SRP3.7.1-SEB1-02 (Supplement 2), stating that either both requirements must be met, or a site-specific evaluation is needed. The applicant also identified a proposed revision to AP1000 DCD Sections 2.5.2.1 and 2.6, to incorporate this information. On the basis that the applicant has identified both essential requirements, RAI-SRP3.7.1-SEB1-02 is technically resolved.

In a revised response dated July 9, 2010, the applicant indicated that a statement will be added to TR-115 to indicate that a comparison of the site-specific Vs profile to the generic HRHF Vs profile is needed in addition to the comparison of the site-specific spectra to the generic HRHF spectra. Confirmatory Item CI-SRP3.7.1-SEB1-02 will track the formal submittal of the AP1000 DCD and TR-115 revisions.

The staff reviewed the description of "Evaluation Methodology" in TR-115, Section 3.0, and noted that the methodology is consistent with the presentation made by the applicant during the April 2007 audit. However, TR-115, Section 3.0, does not include any of the quantitative information presented at the audit to demonstrate the implementation of the approach. In RAI-SRP3.7.1-SEB1-03, the staff requested that the applicant make available for audit, a detailed report of numerical results that demonstrate the implementation specifically for the AP1000. During the May 2008 audit, the staff reviewed the applicant's report, which documents the implementation of the methodology, and concluded that it is consistent with the presentation made to the staff during the April 2007 audit and the staff's ISG on incoherency. Initially, the staff considered RAI-SRP3.7.1-SEB1-03 to be resolved. However, the applicant subsequently revised the ACS-[ ] NI20 model used for the HRHF analysis, in order to correct modeling errors identified by the staff during its confirmatory analysis effort. The staff identified the review of the revised analysis results as Open Item OI-SRP3.7.1-SEB1-03 in the SER with open items. During the June 14-18, 2010 audit, the staff reviewed the revised NI20 [ ] model (in calculation report, [ ]) to ensure that modeling corrections had been addressed. The staff verified that the [ ] model properly represented the actual AP1000 NI structural features. The staff also confirmed that seismic motion incoherency was implemented in accordance with the staff's ISG. Based on the staff's audit of [ ], RAI-SRP3.7.1-SEB1-03 and the associated open item are resolved.

The staff reviewed the details of the "Screening Criteria" in Section 4.0 of TR-115. The applicant lists four screening criteria used to select SSCs for detailed evaluation. Based on the screening criteria, it was not clear to the staff why the containment structure is not identified for detailed comparison of the CSDRS and the HRHF responses. In RAI-SRP3.7.1-SEB1-04, the staff requested that the applicant either include a detailed comparison for the containment structure in Section 6.1, or describe in detail its technical basis for excluding the containment structure.

In its response dated April 25, 2008, the applicant stated that the steel containment structure was not chosen for evaluation since it does not meet the criterion of significant modal response within the region of high frequency amplification. The applicant stated that the dominant frequencies for horizontal response are below 10 Hz, and the dominant mode in the vertical direction is below 20 Hz, which are not in the region where the HRHF spectra exceed the AP1000 CSDRS; and that over 75 percent of the containment structure mass participates in modes below the frequency where the HRHF spectra exceed the CSDRS. The staff evaluated the above response and initially concluded that the basis for excluding the containment shell was adequately described. However, the staff subsequently noted that AP1000 DCD Revisions 16 and 17, Section 3G.2.1.3, identifies high frequency modes (20-30 Hz) in the upper closure dome of the steel containment. Since high frequency modes in the upper closure dome were not addressed in TR-115, or in the initial RAI response, the staff requested that the applicant submit a supplemental RAI response justifying why these modes in the upper closure dome would not be excited by the HRHF ground spectra.

In its supplemental response dated September 12, 2008, the applicant stated that the seismic response spectra in the vicinity of the polar crane (~68 m (~224 ft) El.) are representative of the seismic response of the upper closure dome, and that the CSDRS spectra envelope exceeds the HRHF FRS at this location. Therefore, the applicant concluded that the closure dome will have lower response due to HRHF excitation than due to CSDRS excitation. The staff found

this response to be inadequate because the results being compared are based on the stick model of the containment structure, which does not include the flexibility of the upper closure dome. The staff requested that the applicant provide information pertinent to addressing the staff's concern.

In its revised response dated May 14, 2009, the applicant stated that the NI20 ACS [ ] analysis for the HRHF ground motion input produced ISRS at the base of the SCV that are completely enveloped by the comparable ISRS produced by the CSDRS ground motion input, across the entire frequency range. The staff reviewed the comparison plots provided in the response, and noted that in this case the HRHF input would not excite the vibration modes in the SCV dome. The staff noted, however, that the applicant needed to confirm this after the HRHF reanalysis was completed. Pending the staff's evaluation of the applicant's revised incoherency analysis results (discussed under RAI-SRP3.7.1-SEB1-03, RAI-SRP3.7.1-SEB1-09, RAI-SRP3.7.1-SEB1-10, and RAI-SRP3.7.1-SEB1-11), this was designated as Open Item OI-SRP3.7.1-SEB1-04 in the SER with open items.

In its revised response dated July 9, 2010, the applicant provided updates to RAI response Figures RAI-SRP3.7.1-SEB-04-1, RAI-SRP3.7.1-SEB-04-2, RAI-SRP3.7.1-SEB-04-3, and RAI-SRP3.7.1-SEB-04-10 that show the corrected spectra comparisons. The staff noted that the CSDRS ISRS still envelope the HRHF ISRS, except for a very minor local exceedance in the Y direction ISRS. Since the input at the base of the SCV is more severe for the CSDRS than for the HRHF spectra, the staff accepts the applicant's decision to screen out the SCV from the HRHF detailed evaluation sample. RAI-SRP3.7.1-SEB1-04, and the associated open item, are resolved.

The staff's reviewed the analytical models described in TR-115, Section 5.0, and noted that the applicant had not adequately justified the applicability of the NI20 model to accurately predict high frequency modes potentially excited by the HRHF ground motion input. In RAI-SRP3.7.1-SEB1-06, the staff requested that the applicant include in Section 5.1 of TR-115 a comparison of frequencies and mode shapes between the more refined NI10 model and the NI20 model, to demonstrate the adequacy of the NI20 model to accurately predict high frequency modes.

In its response dated April 25, 2008, the applicant stated that at the December 20, 2007, meeting between the staff and industry related to the high frequency seismic events, it was agreed that a maximum analysis frequency of 50 Hz would be sufficient to transmit the high frequency response through the model. The applicant further stated that using the NI20 model (mesh size of 6.1 m (20 ft), and the shortest wavelength of 42.1 m (138 ft), there are close to 7 nodes per wavelength, to transmit the high frequency through the finite elements; and stated that it is not necessary to include in Section 5.1 a comparison of frequencies and mode shapes between the NI10 and NI20 models.

During the May 2008 audit, the staff noted that NUREG-0800 Section 3.7.2 (Revision 3, March 2007) identifies the staff's expectations for demonstrating adequacy of the element refinement to accurately simulate behavior at the highest frequency of interest, and requested the applicant to submit additional information to demonstrate the adequacy of the NI20 model. The applicant submitted supplemental responses in September 2008, January 2009, and

June 2009. The staff reviewed these supplemental responses and concluded that none of the information submitted directly addressed the staff's initial RAI question.

As a result of the inadequate responses from the applicant, the staff initiated an independent confirmatory analysis effort in June 2009, to study the modal properties of both the NI10 and NI20 models and compare the two models up to 50 Hz. Based on this effort, the staff concluded that the overall building response is adequately represented in the NI20 model, up to 50 Hz. However, local panel vibration modes of walls, floors, and ceilings, up to 50 Hz, are not necessarily modeled with sufficient refinement in the NI20 model. The staff's concern is that, if the NI20 model cannot accurately predict the amplified response of flexible regions up to 50 Hz, then any HRHF high frequency exceedances of the design ISRS (based on the CSDRS) cannot be accurately predicted. Therefore, the staff requested the applicant to review the NI20 model to determine which wall, floor, and ceiling panels are not modeled with sufficient refinement, and to address how this affects the structural design loads and the ISRS, for the HRHF ground spectra input. This was identified as Open Item OI-SRP3.7.1-SEB1-06 in the SER with open items.

In its revised response dated July 27, 2010, the applicant stated that the procedure for addressing the out-of-plane response of flexible regions was the same as that described in its revised response (July 9, 2010) to RAI-TR03-032. The applicant used the NI05 model to identify flexible regions where the out-of-plane response may occur at frequencies less than 50 Hz. The staff's review of the applicant's July 9, 2010 response RAI-TR03-032 is in Section 3.7.2.4.1 of this SER. Based on its review, the staff finds the applicant's method for identifying flexible regions (below 50 Hz) and modifying the ISRS to be acceptable. By using the mesh refinement of the NI05 model, the applicant was able to locate and evaluate flexible regions of the NI structures that were inadequately modeled in the less refined NI20 model. The applicant identified proposed changes to TR-115 to document the new procedure. Therefore, RAI-SRP3.7.1-SEB1-06 and the associated open item are technically resolved. Confirmatory Item CI-SRP3.7.1-SEB1-06 will track the formal submittal of the TR-115 revision.

In its review of the NI10 and NI20 spectral comparisons in Section 5.1, the staff noted that the locations presented showed no significant amplification in the high frequency range. In RAI-SRP3.7.1-SEB1-08, the staff requested that the applicant include in Section 5.1, NI10 versus NI20 comparisons at locations and in directions where there is significant amplification at high frequency.

In its response dated September 12, 2008, the applicant stated that Figures 5.1-4 and 5.1-5 would be added to Section 5.1 of TR-115 to show the locations and response spectra at additional locations. The staff reviewed the supplemental response and found that the two added locations exhibit more significant response in the high frequency region than the three original locations. Significant spectral amplification in X and Y is generally in the 10-20 Hz range, with one Y-direction peak in the 20-30 Hz range. Significant spectral amplification in Z-direction is generally in the 20-30 Hz range.

The staff noted, however, that the comparisons presented did not demonstrate any consistent pattern of correlation among the three models ([ ] NI10, [ ] NI20, and [ ] NI20). In two of the horizontal comparisons, there are significant differences in the 7-8 Hz range, where excellent correlation would be expected. The staff concluded that although the applicant's

response addressed the information request, there was no discussion of the anomalies in the comparisons. The staff was concerned that the applicant had not conducted a sufficient assessment of these results before submitting them. Therefore, the staff issued RAI-SRP3.7.1-SEB1-08, Supplement 1, describing the anomalies and requesting the applicant to review and comment on them.

In it supplemental response dated February 24, 2009, the applicant stated that the results presented were obtained from different models (NI10 and NI20) and different technologies ([ ] - time domain solution, and [ ] - frequency domain solution), and that this can result in the differences identified. The applicant stated that the response spectra show:

- In general the shapes of the response spectra are similar.
- The NI20 model has higher response than the NI10 model.
- [ ] analyses are conservative.

The staff determined that the applicant had not addressed the specific questions posed by the staff, and discussed this with the applicant during the April 2009 audit. The applicant agreed to provide additional information to explain the inconsistencies noted by the staff.

In a letter dated June 3, 2009, the applicant submitted a supplemental response to this RAI, explaining that the inconsistent results reported in the Z direction between nodes 2247 and 2078 was due to modeling differences between the NI10 and NI20 models. The staff reviewed the additional information, and concluded that the explanation is plausible, but not conclusive. The staff determined that resolution of this RAI would need to be deferred until the staff had completed its independent confirmatory analysis program. This was identified as Open Item OI-SRP3.7.1-SEB1-08 in the SER with open items.

The results of the staff's confirmatory analysis of the NI20 [ ] model are described under OI-SRP3.7.1-SEB1-09, OI-SRP3.7.1-SEB1-10, and OI-SRP3.7.1-SEB1-11. The staff identified errors in the applicant's NI20 [ ] model, which required the applicant to perform a reanalysis of all [ ] runs. During the June 14-18, 2010 audit, the staff reviewed the revised NI20 [ ] model and results (in calculation report, [ ]). The staff verified that the revised [ ] model properly represented the actual AP1000 NI structural features.

In its revised response dated July 9, 2010, the applicant indicated that the differences in response between the southeast and northeast corners of the auxiliary building, as depicted in corrected TR-115 Figures 5.1-7 and 5.1-8, are due to local differences in geometry between the NI10 and NI20 models, and also due to differences in the seismic ISRS at the base of the auxiliary building, between [ ] and [ ]. The applicant also identified a proposed revision to TR-115. The staff determined that the applicant's response is acceptable, on the basis that these results are not design-basis results, but are only intended to demonstrate dynamic similarity between the three models ([ ] NI10, [ ] NI20, and [ ] NI20). Also, as discussed under RAI-SRP3.7.1-SEB1-06, there are local regions where NI20 does not possess the necessary model refinement to represent modal behavior up to 50 Hz. In these areas, the applicant is relying on the more refined NI05 model to develop HRHF ISRS. Therefore, the staff considers RAI-SRP3.7.1-SEB1-08, and the associated open item to be technically resolved. Confirmatory Item CI-SRP3.7.1-SEB1-08 will track the formal submittal of the TR-115 revision.

The staff reviewed the HRHF ISRS presented in TR-115, Section 5, and issued three related RAIs. RAI-SRP3.7.1-SEB1-09, RAI-SRP3.7.1-SEB1-10, and RAI-SRP3.7.1-SEB1-11 requested the applicant provide clarification and explanation of in-structure response reductions and apparent inconsistencies in the presented results. The significant issues raised by the staff and the applicant's responses follow.

(1) The staff noted that the spectral acceleration ratio of coherent motion to incoherent motion is as high as 3, and a ratio of 2 is fairly common. The staff requested that the applicant provide the detailed technical basis for concluding that the calculated reductions are reasonable, and consistent with the ISG on this subject; and also to identify whether any independent peer review of this result had been performed, considering it is a first-time application of this technology.

(2) The staff noted that spectral acceleration reductions are indicated at frequencies as low as 6-10 Hz. The staff requested that the applicant provide the detailed technical basis for concluding that the calculated reductions at a low frequency are reasonable, and consistent with the ISG on this subject; and also to identify whether any independent peer review of this result had been performed, considering it is a first-time application of this technology.

(3) The staff noted that even when the beneficial effects of incoherency are included, there are high frequency exceedances at a number of the sample locations evaluated. However, the applicant apparently has concluded that the worst-case exceedances have been determined, without expanding the sample size and evaluating additional locations. The staff requested that the applicant provide a detailed technical basis for concluding that the seismic response of AP1000 SSCs to the defined HRHF ground spectra input is enveloped by the response at the selected sample locations.

(4) The staff reviewed the ISRS for the containment operating floor, east side, El. 40.9 m (134.25 ft) (Node 2136), and for the containment operating floor, west side, El. 40.9 m (134.25 ft) (Node 2170), in TR-115, Revision 1, Figure 5.2-2. The staff observed that the east side and west side Y-direction spectra are very similar. However, the east side and west side X-direction spectra and the east side and west side Z-direction spectra are very different, for both the HRHF-coherent and HRHF-incoherent cases.

Location	Direction	HRHF-coherent	HRHF-incoherent
East Side	X	1.6g (20 Hz)	1.05g (20 Hz)
West Side	X	3.5g (13 Hz)	2.8g (13 Hz)
East Side	Y	3.5g (16 Hz)	1.95g (16 Hz)
West Side	Y	3.7g (16 Hz)	2.05g (16 Hz)
East Side	Z	1.9g (40-50 Hz)	0.65g (40-50 Hz)
West Side	Z	3.2g (30 Hz)	1.7g (30 Hz)

The staff could not determine a rational explanation for this behavior, and requested the applicant to provide a detailed technical explanation for these apparently inconsistent results.

In a letter dated February 4, 2009, the applicant provided the following response:

(1) [ ]-Simulation incoherency approach used to generate the seismic response spectra is in accordance with Section 4, Subsection 1.0 of "Interim staff Guidance (ISG) on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combine License Applications," supplements to Section 3.7.1, "Seismic Design Parameters," of NUREG-0800. In generating the seismic response spectra, the applicant made no changes to the accepted industry methodology. The technical basis for incoherence is discussed in EPRI Report 1012966, "Effect of Seismic Wave Incoherence on Foundation and Building Response," December 2005. Similar results were shown in Figure 6-1 to 6-11 of EPRI Report 1012966. Figure 6-12 showed 5-fold reduction at 50 Hz.

(2) See (1) above. Figure 6-6 of EPRI Report 1012966 showed the similar reduction at 10 Hz.

(3) The applicant had agreed to evaluate a representative sample of SSCs located in areas that are subject to high frequency response, and have frequency content in the high frequency region, to confirm that high frequency seismic input is not damaging, and to demonstrate that normal design practices using the CSDRS result in an AP1000 design that is safer and more conservative. This evaluation is reported in TR-115. The SSCs selected based on the screening criteria are sufficient to demonstrate that high frequency seismic events are not damaging. There may be spectra that have higher exceedances; however, safety-related equipment may not be located in these locations, SSCs located in these areas may not have high frequency response, and further the evaluation performed demonstrates that the HRHF seismic event is not damaging and there is margin between the CSDRS and HRHF response. The applicant's evaluation approach is in compliance with Section 4, Subsection 3.0 and 4.0 of the "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications."

(4) Figure RAI-SRP3.7.1-09-C (in the response) shows the location of nodes 2136 and 2170. Node 2170 is surrounded by a large semi-circle IRWST water tank while node 2136 is surrounded by concrete structure floor and steam generator compartment wall. Node 2136 showed more interaction in X and Z direction between the CISs. The responses of both nodes in Y direction are similar because of less structure interaction between the steam generator compartment wall and other concrete structure. The differences between coherent and incoherent responses are justified in (1) and (2) above.

The staff reviewed the applicant's responses to the supplemental information request, and determined that the responses to (1) and (2) were unacceptable, because the applicant referenced an EPRI report that is not referenced in ISG-1. The applicant needed to confirm that it used the specific reports referenced in the ISG, dated May 19, 2008. If this is not the case, then the applicant would need to perform new analyses that are consistent with the ISG

approved methods. Also, the applicant had to confirm that the results questioned by the staff in (1) and (2) are consistent with results presented in TRs that the staff has accepted.

The staff discussed this RAI response with the applicant during the April 2009 audit. The staff determined that the best course of action to resolve the remaining staff concerns on Items (3) and (4) was to conduct independent confirmatory analyses. To support this effort, the applicant agreed to submit the [ ] NI20 model used in its incoherency analyses to the staff. The staff also requested the applicant to conduct several parametric analyses, using a simplified AP1000 model from the EPRI studies and varying the basemat dimensions and properties of the foundation media.

Confirmatory Analysis

To support the staff's review of the applicant's responses to RAI-SRP3.7.1-SEB1-09, RAI-SRP3.7.1-SEB1-10, and RAI-SRP3.7.1-SEB1-11, the staff initiated a confirmatory analysis effort in May 2009. The applicant provided the staff with the seismic analysis models ([ ] NI20 and [ ] NI20), so that an independent check of modeling assumptions could be performed. In the confirmatory analysis effort, the staff identified several key findings:

1. The staff identified several modeling errors in the applicant's [ ] NI20 model. The errors related to the end-release assumptions for certain beam elements and their effect on over-constraining the global [ ] model. In addition, there were several foundation nodes on the NI basemat that were not identified as [ ] interface nodes. It was not clear to what extent these modeling errors might affect ISRS as well as the ZPA values used for structural design. The staff informed the applicant, during the August 2009, audit in Cranberry, Pennsylvania, of these errors and that the errors are likely to affect the results presented in TR-115 and TR-03. The applicant agreed to submit revised results for all prior [ ] analyses reported in TR-115 and TR-03.

2. The staff studied the adequacy of the NI20 model refinement to reasonably predict all vibration modes up to 50 Hz, as specified in the ISG. The conclusion is that there are local regions (i.e., floor, wall, and roof slabs) where the refinement is not sufficient to pick up a local 50 Hz vibration mode. Therefore, the ISRS may not be accurate in these areas. In RAI-TR03-032 and RAI-SRP3.7.1-SEB1-06, the staff requested the applicant to review the NI20 model, locate all such local areas, determine whether there are mounted systems and components in these areas, and describe how the appropriate ISRS will be developed for these areas.

3. The staff compared results between ACS [ ] and the latest version of [ ] 2000, for the AP1000 NI20 model and HRHF ground motion, with and without incoherency effects. There are significant reductions in the low frequency region of the ISRS when incoherency effects are included. The staff found that the low frequency reductions were not consistent with EPRI calculations referenced in ISG-1. The staff's review of the applicant's use of incoherency is discussed below.

Use of Incoherency

The staff focused its review of the applicant's use of spatial incoherency by requesting the applicant (RAI-SRP3.7.1-SEB1-10) to provide comparisons of ISRS using both coherent and incoherent input motion. In response to RAI-SRP3.7.1-SEB1-10, the applicant provided response spectra comparisons at several locations on the NI:

A. Top of the shield building (El. 99.8 m (327.4 ft))
B. East side of the containment operating floor (El. 40.9 m (134.25 ft))
C. West side of the containment operating floor (El. 40.9 m (134.25 ft))
D. Shield building, northeast corner (El. 40.9 m (134.5 ft))
E. Shield building, at fuel building roof (El. 54.7 m (179.6 ft))
F. Reactor coolant pump (El. 30.2 m (99.0 ft))

For the purpose of comparing the applicant's results to previous EPRI calculations, the staff reviewed the response spectra comparisons, and developed approximate ratios of incoherent to coherent motion in the low and high frequency ranges. These comparisons are provided in SER Table 3.7-1. The applicant also stated that the Abrahamson Hard-Rock Coherency Model (2007), as incorporated into ACS-[ ], was used to perform SSI calculations. The staff finds that the applicant's use of the 2007 Abrahamson Hard-Rock coherency model is consistent with staff guidance (i.e., ISG-1).

Table 3.7-1. Incoherent Versus Coherent Response (Approximate)

Building Location	Direction	Incoherent/Coherent Response Ratio	
		0-10 Hz	10-50 Hz
Top of the shield building	X	0.90	0.75
	Y	0.95	0.85
	Z	0.65	0.90
East side of the containment operating floor	X	0.90	0.75
	Y	0.90	0.70
	Z	0.90	0.55
West side of the containment operating floor	X	0.90	0.85
	Y	0.85	0.75
	Z	0.90	0.50
Shield building, Northeast corner	X	0.85	0.70
	Y	0.95	0.75
	Z	0.80	0.65
Shield building, at fuel building roof	X	0.85	0.75
	Y	0.80	0.75
	Z	0.80	0.60
Reactor coolant pump	X	0.90	0.90
	Y	0.80	0.95

Table 3.7-1. Incoherent Versus Coherent Response (Approximate)

Building Location	Direction	Incoherent/Coherent Response Ratio	
		0-10 Hz	10-50 Hz
	Z	0.75	0.85

The results shown in SER Table 3.7-1 indicate that low frequency reductions range from 5-35 percent. The locations of the most significant response reductions are at the top of the shield building and at the reactor coolant pump, with approximately 25-35 percent reductions in the 0-10 Hz range.

High-frequency response reductions range from 5-50 percent. The locations of the most significant high-frequency reductions are at the east and west sides of the containment operating floor, in the vertical direction, and the shield building (at fuel building roof), in the Y direction. Approximate reduction of 45-50 percent in the 10-50 Hz range was observed at these locations.

The staff also reviewed spectral response comparisons for several nodes on the basemat. These basemat nodes exhibited similar reductions in response both in the low and high frequency ranges. The staff finds that the high-frequency response predictions are reasonable based on comparisons with similar calculations performed by EPRI (TR-1015111, 2007) using more simplified structural models. However, the staff finds that the applicant's low-frequency response reductions, in excess of 30 percent, to be unsupported by the EPRI calculations. To address this concern, in RAI-SRP3.7.1-SEB1-11, the staff requested the applicant provide justification for the significant reductions in a low frequency response.

In its response, the applicant stated that the low frequency reductions were due to the use of the 2007, HR coherency function itself, which can have a 50 percent reduction at 50 m in the 2-5 Hz range. The staff found the applicant's justification inadequate because the applicant referenced EPRI calculations (TR-1015111, 2007, Chapter 5), which are based on a soil coherency model that is not applicable to HR sites. The staff notes that Appendix B of the same EPRI report includes results using the approved 2007 coherency function and serves as the staff's basis for comparison.

The staff investigated the applicant's low-frequency response predictions. With the intent of reducing computational effort, the staff developed a simplified FEM of the AP1000 NI. This reduced model was then used for SSI analysis using the ACS-[ ] and [ ]-square root of the sum of the square (SRSS) codes. The simplified SSI model had dynamic response characteristics similar to those of the applicant's more detailed NI model, for frequencies below about 15 Hz. The dynamic response of the simplified model was confirmed by comparing fixed-base TFs at several locations to the more detailed AP1000 NI model. A transfer function is defined as a frequency-dependent function of SSI amplification due to a unit input motion. Further, for incoherent analysis using both analytic formulations recognized by the ISG, the confirmatory analyses used the same 2007 Abrahamson coherency function that the applicant referenced, as well as the applicant's HRHF input motion.

The staff performed SSI analyses using the simplified model for both coherent and incoherent motion. The goal of this analysis was to determine if the low frequency reductions of ISRS seen in the applicant's analysis could be duplicated with [ ]. This analysis also used the same HR site and HRHF input motion provided by the applicant.

The SSI analysis results using [ ]-SRSS for the simple NI model, as well as the full NI20 FEM with HRHF input, indicate negligible reductions in ISRS in the low frequency range due to incoherency effects. SSI TFs of the simplified model from both [ ]-SRSS and ACS-[ ] show negligible reductions in the low frequency range (below 10 Hz). In addition, 5 percent damped ISRS from [ ]-SRSS analysis of the NI20 model exhibit only negligible reductions at low frequency.

Based on the review of the applicant's results and the staff's independent confirmatory analysis efforts, the staff concluded that the applicant's predictions of in-structure response in the low frequency range were not consistent with EPRI's calculations and the staff's confirmatory calculations. The staff also noted that the applicant's high-frequency incoherent results cannot be considered acceptable if low frequency results cannot be validated. These issues are identified as Open Items OI-SRP3.7.1-SEB1-09, OI-SRP3.7.1-SEB1-10, and OI-SRP3.7.1-SEB1-11.

During the June 14-18, 2010 audit, staff reviewed the revised NI20 [ ] model (in calculation report, [ ]) to ensure that modeling corrections had been addressed. The staff verified that the [ ] model was properly transferring bending moments at the beam (or shell) connections with solid elements. In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-SRP3.7.1-SEB1-11. The applicant indicated that [ ] modeling corrections (e.g., beam element and shell element connections to solid elements) had been addressed and the reanalysis had been performed.

The applicant provided ACS [ ] results for the corrected NI20 model. Using the incoherency option in ACS [ ], the applicant developed ISRS results for 25 simulations (with and without phase adjustment) for the AP1000 NI six key locations (shown in Figures RAI-SRP3.7.1-SEB1-11-50 through RAI-SRP3.7.1-SEB1-11-67 of the response). The staff reviewed these comparisons and finds that while there are some differences between the original HRHF results and the corrected results (with phase adjustment), the original HRHF results are generally conservative.

The applicant also provided ISRS comparisons (coherent and incoherent) at the four corners and center of the NI basemat (shown in Figures RAI-SRP3.7.1-SEB1-11-68 through RAI-SRP3.7.1-SEB1-11-82 of the response). The applicant stated that these analyses incorporate a phasing correction, which no longer results in significant low-frequency reductions. The staff reviewed these comparisons and finds that there are minimal (<10 percent) ISRS reductions below 10 Hz for the locations presented.

Based on review of the applicant's corrected NI20 [ ] model and the new HRHF results, the staff finds that the applicant has properly implemented modeling corrections, and the ISRS show negligible reductions due to incoherency below 10 Hz. On the basis of these findings, RAI-SRP3.7.1-SEB1-11 and associated open item are resolved.

In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-SRP3.7.1-SEB1-09. In response to a request from the staff, the applicant identified the following proposed addition to TR-115, Section 5.2:

> The exceedances of CSDRS-based ISRS by HRHF-based ISRS are addressed as part of the sampling evaluation documented in this report to confirm that high frequency input has marginal effect on equivalent piping, and structures qualified by analysis for the AP1000 CSDRS.

The applicant had previously addressed issue (4) described above in its February 9, 2009, response, by providing Figure RAI-SRP3.7.1-09-C in the response, which shows the location of nodes 2136 and 2170, and stated that node 2170 is surrounded by a large semi-circular IRWST water tank, while node 2136 is surrounded by concrete structure floor and steam generator compartment wall. The applicant noted that node 2136 showed more interaction in X and Z direction between the CISs. The responses of both nodes in Y direction are similar because of less structure interaction between the steam generator compartment wall and other concrete structure. Prior to the staff's confirmatory analysis, and the applicant's reanalysis after correction of modeling errors, the staff had reserved judgment on the applicant's explanation. With the resolution of RAI-SRP3.7.1-SEB1-10 and RAI-SRP3.7.1-SEB1-11, the staff has concluded that the applicant's explanation for the differences is viable. Therefore, RAI-SRP3.7.1-SEB1-09, and the associated open item, are technically resolved. Confirmatory Item CI-SRP3.7.1-SEB1-09 will track the formal submittal of the TR-115 revision.

Acceptability of ISRS Reductions

In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-SRP3.7.1-SEB1-10, which provided the reanalysis for seismic response, using the corrected NI20 model. In Figures RAI-SRP3.7.1-SEB1-10-1 to RAI-SRP3.7.1-SEB1-10-21 of the response, the applicant provided incoherent and coherent ISRS comparisons. The applicant stated that some ratios of incoherent-to-coherent response are shown to be less than 0.5. To justify this level of reduction, the applicant used the EPRI AP1000 stick model to compare ISRS reductions to the 3D AP1000 model. Three cases were analyzed: EPRI stick model with EPRI soil profile and EPRI time history; EPRI stick model with EPRI soil profile and HRHF time history; EPRI stick model with HRHF soil profile; and HRHF time history input. The results of these analyses are shown in Figures RAI-SRP3.7.1-SEB1-10-22 to RAI-SRP3.7.1-SEB1-10-33 of the response. The results showed that a larger foundation will have a larger reduction in response due to incoherency effects. The results for the top-of-CIS show reductions of the magnitude seen in the NI20 results (approximately 50 percent reduction). The top of the SCV and top the shield building also show similar results. Figures RAI-SRP3.7.1-SEB1-10-34 and RAI-SRP3.7.1-SEB1-10-35 of the response show a comparison of the basemat response of the NI20 model and the EPRI stick models. The comparison shows that the reductions due to incoherency are similar in magnitude.

The staff reviewed the applicant's comparison of incoherent and coherent results and finds the results similar to those developed independently (SER Table 3.7-1). Based on the similar ISRS reductions of the AP1000 to the EPRI calculations (which are referenced in the ISG), the staff finds the applicant's reductions due to the use of incoherency to be acceptable. Therefore, RAI-SRP3.7.1-SEB1-10, and the associated open item are resolved.

Evaluation of Structures for HRHF Loading

During the April 2007 audit, the applicant presented structural response comparisons between CSDRS loading and HRHF loading. The staff obtained clarification from the applicant that the HRHF results assumed coherent motion. However, the staff noted that TR-115, Section 6.1, did not identify whether the structural response comparisons in Tables 6.1-1 through 6.1-6, between CSDRS loading and HRHF loading, assumed coherent motion or incoherent motion. In RAI-SRP3.7.1-SEB1-12, the staff requested that the applicant clearly define how it calculated the HRHF structural loads presented in TR-115, Tables 6.1-1 through 6.1-6.

In a letter dated April 25, 2008, the applicant stated that the HRHF member forces provided in Tables 6.1-1 through 6.1-6 are based on incoherency. The incoherent member forces are averaged from 25 independent Monte Carlo runs done with [ ] and multiplied by the element thickness to form the member forces presented.

The staff also requested, in RAI-SRP3.7.1-SEB1-13, that the applicant provide additional comparison results in Tables 6.1-1 through 6.1-6, based on use of the HRHF ground motion without considering reduction for incoherency, similar to the results presented in April 2007. In a letter dated April 25, 2008, the applicant provided the requested comparisons between the coherent and incoherent results in a set of tables designated RAI-SRP3.7.1-SEB1-13-01 to RAI-SRP3.7.1-SEB1-13-01-6. The applicant also noted that it had identified inconsistencies in the HRHF incoherent results tabulated in TR-115, and referred to its response to RAI-SRP3.7.1-SEB1-14.

During review of TR-115, Tables 6.1-1 through 6.1-6, the staff had noted several erratic patterns of differences between the CSDRS results and the HRHF results. In RAI-SRP3.7.1-SEB1-14, the staff requested that the applicant review the tabulated results in Tables 6.1-1 through 6.1-6, and provide a technical explanation for all patterns of differences that the applicant determined to be in need of further review.

In a letter dated April 25, 2008, the applicant stated that it had reviewed the tabulated results in Tables 6.1-1 through 6.1-6 and concluded that there were inconsistencies in the tabulated results. These inconsistencies were corrected; the revised tables were included in the RAI response, and also identified for inclusion in TR-115, Revision 1. The applicant stated that the conclusions in Section 6.1 remain unchanged. During the May 2008, audit, the staff discussed these three RAI responses with the applicant. The expanded and corrected results included in the response to RAI-SRP3.7.1-SEB1-13 show that the HRHF coherent results are enveloped by the CSDRS results. Therefore, the staff concluded that structures designed to the CSDRS input are also adequately designed for the HRHF input. The staff also confirmed that the corrected tables were included in TR-115, Revision 1. On this basis, RAI-SRP3.7.1-SEB1-12, RAI-SRP3.7.1-SEB1-13, and RAI-SRP3.7.1-SEB1-14 were resolved.

3.7.2.5 Development of Floor Response Spectra

In AP1000 DCD, Revision 17, Section 3.7.2.5, the applicant stated that design FRS are generated according to RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," Revision 1. The seismic FRS is

computed using time-history responses determined from the NI seismic analyses. The time-history responses for the HR condition are determined from a mode superposition time history analysis using computer program [ ]. The time-history responses for the FR and soil conditions are determined from a complex frequency response analysis using the computer program, [ ]. FRS for damping values equal to 2, 3, 4, 5, 7, 10, and 20 percent of critical damping are computed at the required locations.

The applicant stated that FRS for the design of subsystems and components are generated by broadening the enveloped nodal response spectra determined for the HR site and soil sites. The spectral peaks are broadened by ±15 percent to account for the variation in the structural frequencies, due to the uncertainties in parameters, such as material and mass properties of the structure and soil, damping values, seismic analysis technique, and the seismic modeling technique. Figure 3.7.2-14 shows the broadening procedure used to generate the design FRS.

The applicant further stated that spectral peaks at frequencies associated with fundamental SSI frequencies are reviewed. If there is a "valley" between peaks due to different soil profiles and not the building modal response, then this valley is filled by extending the broadening of the lower peak horizontally until it meets the broadened upper peak. The SSE FRS for 5 percent damping, at representative locations of the coupled ASBs, the SCV, and the CIS, are presented in AP1000 DCD, Revision 17, Appendix 3G.

Based on its review of AP1000 DCD, Revision 17, Section 3.7.2.5, and the related information in Appendix 3G, the staff concluded that the applicant's approach for enveloping the multiple site responses, and filling any "valley" in the envelope attributable to soil response, is consistent with current staff guidance, and is acceptable.

3.7.2.6 Three Components of Earthquake Motion

In AP1000 DCD Section 3.7.2.6, the applicant stated that seismic system analyses are performed considering the simultaneous occurrences of the two horizontal and the vertical components of earthquake. In mode superposition time-history analyses using the computer program, [ ], the three components of earthquakes motions are applied either simultaneously or separately. In the [ ] analyses with three component earthquake motion applied simultaneously, the effect of the three components of earthquake motion is included within the analytical procedure so that further combination is not necessary. In analyses where the earthquake components are applied separately, the three components of earthquake motion are combined using one of the following methods:

- For seismic analyses with the statistically independent earthquake components applied separately, the time-history responses from the three earthquake components are combined algebraically at each time step to obtain the combined response time-history. This method is used in the [ ] analyses.

- The peak responses due to the three earthquake components from the response spectrum and equivalent static analyses are combined using the SRSS method.

- The peak responses due to the three earthquake components are combined directly, using the assumption that when the peak response from one component occurs, the

responses from the other two components are 40 percent of the peak (100 percent-40 percent-40 percent method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus), are considered. This method is used in the NI basemat analyses, the CV analyses and the shield building roof analyses.

The applicant further stated that the CV is analyzed using axisymmetric FEMs. These axisymmetric building structures are analyzed for one horizontal seismic input from any horizontal direction and one vertical earthquake component. Responses are combined by either the SRSS method or by a modified 100 percent-40 percent-40 percent method in which one component is taken at 100 percent of its maximum value and the other is taken at 40 percent of its maximum value.

The applicant stated that a summary of the dynamic analyses performed and the combination techniques used is presented in AP1000 DCD Appendix 3G. In Appendix 3G.4.3.1, the applicant indicated that for RSA, the SRSS method is used to combine the spatial components, in accordance with Section 2.1 of RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 2.

The staff reviewed the update to AP1000 DCD Section 3.7.2.6, and related information in Appendix 3G, and concluded that: (1) algebraic combination at each time step is consistent with standard practice and the staff guidance for time history analyses using three statistically independent inputs, including SSI analyses using [ ], and is acceptable; and (2) use of the SRSS combination is consistent with standard practice and the staff guidance for RSA, equivalent static analysis, and time history analysis when the three inputs are not statistically independent, and is acceptable.

In NUREG-1793 for the AP1000 DCD, Revision 15, the staff had accepted the use of the 100-40-40 method for combining the responses due to the three components of earthquake motion, when the equivalent static acceleration method is used. In July 2006, the staff issued RG 1.92, Revision 2, which included guidance on implementation of the 100-40-40 method. After the submittal of the AP1000 DCD, Revision 17, the applicant identified significant design changes to the roof of the shield building, which is analyzed for seismic response using equivalent static analysis and the 100-40-40 combination method. In addition, equivalent static analysis and the 100-40-40 combination method are used for seismic evaluation of the containment structure and the basemat. Therefore, the staff inquired whether the applicant had implemented the 100-40-40 method in accordance with the guidance provided in RG 1.92, Revision 2. The staff's safety concern was that improper implementation of the 100-40-40 combination method may result in unconservative estimates of seismic demands. This issue was addressed by Open Item OI-TR85-SEB1-27. This open item has been resolved, and the staff has accepted the applicant's implementation of the 100-40-40 method, based on comparison of the applicant's results to results using the SRSS combination method. See Section 3.8.4.1.1.3.4 of this SER for the staff's detailed assessment.

3.7.2.7 Combination of Modal Responses

In AP1000 DCD, Revision 17, Section 3.7.2.7, the applicant stated that the modal responses in a RSA are combined using the grouping method shown in Section C of RG 1.92, Revision 1,

and when high frequency effects are significant, they are included using the procedure given in Appendix A to NUREG-0800 Section 3.7.2. The applicant further stated that in the fixed base mode superposition time history analysis of the HR site, the total seismic response is obtained by superposing the modal responses within the analytical procedure so that further combination is not necessary. This is unchanged from the AP1000 DCD, Revision 15.

A summary of the dynamic analyses performed and the combination methods used are presented in AP1000 DCD, Revision 17, Appendix 3G. In paragraph 3G.4.3.1, the applicant indicated that the RSA is conducted in accordance with Sections 1.1.3, 1.3.2, 1.4.2, and 1.5.2 of RG 1.92, Revision 2. The staff noted that the applicant's use of the guidance in RG 1.92, Revision 2, for combination of modal responses in RSA, is acceptable because it is consistent with the latest staff guidance on this subject.

However, the staff could not determine whether the applicant's mode superposition time history analyses adequately account for the residual rigid response associated with natural vibration modes with frequencies higher than the input spectrum ZPA frequency. RG 1.92, Revision 2, incorporates more recent research findings with respect to modal response combination methods and the treatment of residual rigid response. It is important to accurately account for the residual rigid response if a nuclear power plant SSC has significant natural vibration modes with frequencies higher than the input spectrum ZPA frequency. Ignoring the residual rigid response in these cases may result in significant underestimation of SSC element forces and moments in the vicinity of supports, as well as underestimation of support forces and moments. In RAI-SRP3.7.1-SEB1-17, part (d), the staff requested the applicant to identify whether the method employed is consistent with or different from the RG 1.92, Revision 2, approach, and to provide the technical basis for the adequacy of any method used that differs from the current staff guidance. The applicant's initial response to the staff's RAI was unsatisfactory. This was identified as Open Item OI-SRP3.7.1-SEB1-17 in the SER with open items.

In its revised response to RAI-SRP3.7.1-SEB1-17, part (d), dated July 27, 2010, the applicant stated that modal superposition time history analysis provides sufficient solution accuracy, without including the residual rigid response, because the modes, which respond beyond the ZPA frequency of the input have no significant contribution to the amplified ISRS. In order to verify the accuracy of the analyses conducted, the applicant performed time history analysis using the NI10 model, with a cutoff frequency of 44 Hz, and an identical time history analysis with additional modes up to 64 Hz for the ASB, and additional modes up to 100 Hz for the CIS. The ISRS comparisons at 5 percent damping are documented in the RAI response at key locations of the ASB and CIS. The applicant provided similar comparisons for key locations in the ASB NI05 model, for 40 Hz and 85 Hz cutoff frequencies. The staff reviewed the comparisons of the ISRS, which showed negligible differences in results between the 2 selected cutoff frequencies. These results support the applicant's position; therefore, the staff concluded that the applicant's implementation of the mode superposition time history analysis method produced sufficiently accurate results, even though it does not formally account for the residual rigid response above the cutoff frequency, as specified in RG 1.92, Revision 2. Therefore, RAI-SRP3.7.1-SEB1-17, part (d), and the associated open item are resolved.

3.7.2.8 Interaction of Noncategory I Structures With Seismic Category I Structures

In AP1000 DCD, Revision 17, Section 3.7.2.8, the applicant described the approach for evaluating the effects of interactions of noncategory I structures with seismic Category I SSCs, and components. The approach identified in the AP1000 DCD, Revision 15, remains unchanged. The evaluation must satisfy one of the following three criteria: (1) collapse of the noncategory I structure will not cause an impact with any seismic Category I SSC; (2) collapse of the noncategory I structure will not impair the intended function of any seismic Category I SSC; or (3) the noncategory I structure is classified as seismic Category II and is analyzed and designed to prevent its collapse under the SSE. The applicant identified three structures adjacent to the AP1000 NI: the annex building, the radwaste building, and the turbine building. There is no change between the AP1000 DCD, Revisions 15 and 17 for the radwaste building. The applicant's evaluation for the radwaste building was previously accepted by the staff.

In the AP1000 DCD, Revision 17, the applicant revised the seismic classification of the annex building. In AP1000 DCD, Revision 15, the entire annex building was classified as seismic Category II. In AP1000 DCD, Revision 17, Section 3.7.2.8.1, the applicant stated that only the portion of the annex building adjacent to the NI is classified as seismic Category II. The applicant stated that the annex building is analyzed for the SSE for the six soil profiles described in AP1000 DCD Section 3.7.1.4 and that for the HR site, a range of soil properties was assumed for the layer above rock at the level of the NI foundation. In RAI-SRP3.7.1-SEB1-15, part (b), the staff requested the applicant to clarify the seismic classification of the remainder of the annex building and confirm that for analysis purposes, the entire annex building has been treated as seismic Category II. If this is not the case, provide the technical basis for not treating it as such.

In its initial response dated February 6, 2009, the applicant stated that as shown in AP1000 DCD Table 3.2-2, the annex building area outlined by columns E-I.1 and 2-13 is classified as seismic Category II. The annex building area outlined by columns A-D and 8-13, as well as column A-G and 13-16 is classified as nonseismic. For design purposes, only the portion identified as seismic Category II is designed following the seismic Category I structures acceptance criteria. The applicant stated that the portions of the annex building classified as nonseismic are not adjacent to the NI, and their collapse will not cause the nonseismic structure to strike a seismic Category I SSC, nor will their collapse impair the integrity of seismic Category I SSCs. The applicant further stated that the nonseismic portion of the annex building is only one story, with roof elevations below 36.7 m (120 ft). If this portion of the annex building failed, it would not cause any failure to the seismic Category II portion that could impair the integrity of the seismic Category I structures.

The staff reviewed the response and determined that additional information was needed about the seismic model used for evaluation of the seismic Category II portion of the annex building; specifically, how the nonseismic portion is incorporated in the model. During the April 2009 audit, the applicant presented pictures of the annex building, showing the seismic Category II and nonseismic portions. The applicant confirmed to the staff that failure of the nonseismic portion is not a safety concern. The applicant stated that the small, single story nonseismic section will be included in the Category I-equivalent seismic analysis of the annex building. The applicant agreed to submit a revision to its earlier response. In a letter dated August 11, 2009, the applicant submitted its revised response, providing the clarifications requested by the staff. Therefore, RAI-SRP3.7.1-SEB1-15, part (b) was resolved.

AP1000 DCD, Revision 17, Section 3.7.2.8.3, describes the design of the turbine building. The applicant revised the description of the turbine building to state that the south end of the turbine building is separated from the rest of the turbine building by a 0.61 m (2 ft) thick RC wall that provides a robust structure around the first bay. This wall isolates the first bay of the turbine building from the general area of the turbine building and from the adjacent yard area. The applicant defined the seismic classification of the turbine building as nonseismic. The staff noted an inconsistency in the turbine building description. AP1000 DCD, Revision 15, Section 3.7.2.8.3, stated "...the major structure of the turbine building is separated from the nuclear island by approximately 18 feet." However, in AP1000 DCD, Revision 17, Section 3.7.2.8.3, this statement and additional descriptive information about the turbine building were deleted. Based on the information in Revision 17, the staff could not determine whether the original classification of the turbine building as nonseismic is still valid.

In RAI-SRP3.7.1-SEB1-15, part (c), the staff requested that the applicant provide the technical basis for not classifying the turbine building as seismic Category II, considering its proximity to the NI and the infeasibility of demonstrating the acceptability of a collapse.

In its initial response dated February 6, 2009, the applicant stated that during the HR certification of the AP1000, the NRC reviewed the classification of the turbine building as a nonseismic structure. The NRC concluded from this review (AP1000 NUREG-1793) "that the method and criteria used for the design of the turbine building will prevent, during a SSE event, the turbine building to jeopardize the safety function of the NI structure, and was therefore acceptable." This conclusion was reached after the applicant agreed to modify the analysis and design requirements to:

- Upgrade the Uniform Building Code (UBC) seismic design from Zone 2A, importance Factor of 1.25, to Zone 3 with an Importance Factor of 1.0 in order to provide margin against collapse during the SSE.
- To use eccentrically braced steel frame structures meeting the requirements given in AP1000 DCD Section 3.7.2.8.3.

The applicant further stated that the turbine building is designed as an eccentrically braced frame structure under the guidance of the UBC and is, by the principle of the code, therefore, designed to deform during the design seismic event rather than collapse. The methods and criteria that were agreed to with the NRC have not changed and are given in AP1000 DCD Section 3.7.2.8.3, Revision 17.

The staff reviewed the response and determined that the applicant had not addressed the staff's question, specifically, the significance of the change from Revision 15 to Revision 17. During the April 2009 audit, the applicant presented pictures of the turbine building, showing: (1) the recent addition of a new seismic Category II portion, which is in close proximity to the NI; and (2) the existing nonseismic portion, which is at a sufficient distance from the NI that failure is not a safety concern.

The applicant stated that any effects of the nonseismic sections of the turbine building on the Category II section of the turbine building will be included in the Category I-equivalent seismic analysis. The applicant agreed to submit a revision to its earlier response. In a letter dated

August 11, 2009, the applicant submitted its revised response, providing the clarifications requested by the staff. Therefore, RAI-SRP3.7.1-SEB1-15, part (c), is technically resolved, and Confirmatory Item CI-SRP3.7.1-SEB1-15 will be resolved upon confirmation of the new seismic Category II portion of the turbine building in a future revision of the AP1000 DCD.

During the April 2009 audit, the staff and the applicant also discussed a related issue, concerning the effects of structure-SSI between the NI and the adjacent Category II structures. These adjacent Category II structures could rest on compacted backfill, with Vs significantly below 1000 fps. The applicant formally submitted its approach in a revised response to RAI-SRP3.7.1-SEB1-15, dated August 11, 2009, which included a discussion of how 2D analysis results will be scaled to simulate 3D behavior in the structure-SSI response. The staff reviewed the applicant's approach for performing structure-SSI analyses of buildings adjacent to the NI, and finds the approach acceptable. However, no analysis results were included in the RAI response. This was identified as Open Item OI-SRP3.7.1-SEB1-15 in the SER with open items.

In a follow-up response submitted July 28, 2010, the applicant provided results of the assessment of structure-SSI for buildings adjacent to the AP1000 NI. The seismic analyses were performed primarily using 2D [ ] models, as shown in Figures RAI-SRP3.7.1-SEB1-15-3 and RAI-SRP3.7.1-SEB1-15-4, included in the response, but the results were corrected by using a 3D-2D effect factor, which was developed using 3D [ ] models of the buildings on rigid foundations, as shown in Figure RAI-SRP3.7.1-SEB1-15-5, included in the response. Three soil cases were analyzed: UBSM, SM, and SS.

The applicant stated that the seismic Category II buildings are designed using the envelope of foundation input response spectra (FIRS) from the AP1000 design basis HR and soil cases, as well as the AP1000 HRHF spectra. The HRHF plant-grade spectra are generated using backfill soil profiles corresponding to Vs of 152.4 m/s (500 fps), 213.36 m/s (700 fps), and 304.8 m/s (1000 fps) at plant grade. The backfill Vs profiles extend from basemat El. 18.4 m (60.5 ft) to grade El. 30.5 m (100 ft). The applicant made a comparison of the resulting forces (axial and shear) and moments and showed, in Figures RAI-SRP3.7.1-SEB1-15-13 and RAI-SRP3.7.1-SEB1-15-14 of the response, that the forces and moments are controlled by the CSDRS demand rather than the HRHF demand. Also in the July 28, 2010, letter response, the applicant proposed to revise AP1000 DCD Section 3.7.2.8.4 to provide screening criteria for the COL applicant for determining whether site-specific analysis is required. If the criteria below are not met, then the COL applicant can perform site-specific analyses to demonstrate that its site-specific seismic Category II foundation seismic response spectra are less than the AP1000 annex building and turbine building first bay generic design envelope foundation spectra. The screening criteria are:

1. The site meets Section 2.5.4.5 AP1000 DCD soil uniformity requirements.

2. For soil sites, the site GMRS is enveloped by the AP1000 CSDRS with soil profiles SS, SM, UBSM, SR, FR, and HR.

3. For HRHF sites, the site GMRS is enveloped by the AP1000 HRHF response spectra with a minimum backfill surface Vs of 500 fps, and a minimum lateral extent of the backfill corresponding to a line extending down from the surface at a one horizontal to

one vertical (1H:1V) slope from the outside footprint limit of the seismic Category II structure.

4. The bearing capacity with appropriate factor of safety is greater than or equal to the bearing demand.

Based on the applicant's structure-SSI analysis results, and the applicant's criteria for requiring site-specific analysis, the staff finds that the applicant's approach to developing seismic demands on seismic Category II structures is acceptable. Consequently, RAI-SRP3.7.1-SEB1-15 and the associated open item are technically resolved. Confirmatory Item CI-SRP3.7.1-SEB1-15(2) will be resolved upon confirmation of the information in a future revision to AP1000 DCD Section 3.7.2.8.4, describing the screening criteria for site-specific analysis.

3.7.2.9 Conclusion

The NRC staff concludes that the revision to the AP1000 DCD continues to support the seismic system analysis for Category I SSCs to meet the applicable NRC regulations for the AP1000 DC.

Aside from the identified confirmatory items discussed above, the revision to the AP1000 certified design provides sufficient information to satisfy the applicable requirements of 10 CFR Part 50, Appendix A, GDC 2, "Design Basis for Protection Against Natural Phenomena"; 10 CFR Part 50, Appendix S; and 10 CFR Part 100, Appendix A, for the seismic design and analysis aspects for Category 1 SSCs to be used in the AP1000 reactor.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each applicant would have to address these issues individually.

3.7.3 Seismic Subsystem Analysis

NUREG-0800 Section 3.7.3, "Seismic Subsystem Analysis," provides guidelines for the staff to use in reviewing issues related to seismic design/analysis of subsystems. This review focused on such subsystems as the miscellaneous steel platforms, steel frame structures, tanks, cable trays and supports, heating, ventilation, and air conditioning (HVAC) ductwork and supports, and conduit and supports. Section 3.7.3, "Seismic Subsystem Analysis" of the AP1000 DCD Revision 15, was accepted in the staff's safety evaluation for the HR site DC, as documented in NUREG-1793. The AP1000 DCD, Revisions 16 and 17, made no changes to AP1000 DCD Section 3.7.3. The staff considers that its previous safety evaluation of AP1000 DCD Section 3.7.3 remains valid.

AP1000 DCD Section 3.7.2 describes the applicant's seismic analysis methods for large atmospheric storage tanks, such as the PCCWST. The PCCWST is located on the top of the shield building and is an integral part of the shield building. The applicant described the modeling and analysis approach for the PCCWST in AP1000 DCD Appendix 3G and TR-03. The staff's review identified the need for additional information. The assessment of this issue is in Section 3.7.2.3 of this SER.

3.7.4 Seismic Instrumentation

This section of NUREG-1793 is unchanged by the AP1000 DCD amendment.

3.7.5 Other Combined License Action Items

This section of NUREG-1793 is unchanged by the AP1000 DCD amendment.

3.7.6 Seismic Design Conclusions

The NRC staff concludes that the proposed amendment to the AP1000 DC, related to the seismic design of Category I SSCs, as described in the evaluation above, are acceptable because they satisfy the applicable requirements of 10 CFR Part 50, Appendix A, GDC 1; 10 CFR Part 50, Appendix S; and 10 CFR Part 100, Appendix A.

The revision to the AP1000 certified design provides sufficient information to satisfy the applicable requirements of 10 CFR Part 50, Appendix A, GDC 1; 10 CFR Part 50, Appendix S; and 10 CFR Part 100, Appendix A for the seismic design and analysis aspects for Category 1 SSCs to be used in the AP1000 reactor.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each applicant would have to address these issues individually.

3.8 Design of Category I Structures

The NRC staff has reviewed the adequacy of the design of Category I structures of the applicant's AP1000 DCD, Revisions 16 and 17 for the standard plant using the guidance provided in Sections 3.8.1, 3.8.2, 3.8.3, 3.8.4 and 3.8.5 of NUREG-0800.

The NRC staff issued NUREG-1793 in September 2004 for AP1000 DCD, Revision 15. Subsequent to the issuance of NUREG-1793, the applicant submitted Revisions 16 and 17 of the AP1000 DCD. Additionally, the following TRs were reviewed:

(1) APP-GW-GLR-005, TR-09, "Containment Vessel Design Adjacent to Large Penetrations"

(2) APP-GW-GLR-045, TR-57, "Nuclear Island: Evaluation of Critical Sections"

(3) APP-GW-GLR-026, TR-44, "New Fuel Rack Design & Structural Analysis"

(4) APP-GW-GLR-033, TR-54, "Spent Fuel Storage Rack Structure/Seismic Analysis"

(5) APP-1200-S3R-003, "Design for the AP1000 Enhanced Shield Building"

(6) APP-GW-GLR-044, TR-85, "Nuclear Island Basemat and Foundation"

(7) APP-GW-GLN-113, TR-113, "AP1000 Containment Vessel Shell Material Specification"

With these revisions, the applicant is seeking to make changes in the following areas: (1) steel containment; (2) concrete and steel internal structures of steel containment; (3) other seismic Category I structures; and (4) foundations. The specific changes in each area are evaluated by the staff using the NUREG-0800 sections identified above.

3.8.1 Concrete Containment

This section is not applicable to the AP1000 design since the AP1000 uses a steel containment.

3.8.2 Steel Containment

Using the regulatory guidance in NUREG-0800 Section 3.8.2, "Steel Containment," the staff reviewed areas relating to steel containments or to other Class MC steel portions of steel/concrete containments. The specific areas of review provided in NUREG-0800 Section 3.8.2 are as follows: (1) description of the containment; (2) applicable codes, standards, and specifications; (3) loads and loading combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, and special construction techniques; (7) testing and inservice surveillance program; (8) inspections, tests, analyses, and acceptance criteria (ITAAC); and (9) COL action items and certification requirements and restrictions. Not all of these areas were applicable to the review of the proposed changes to AP1000 Section 3.8.2 and the following SER sections provide the staff's evaluation for the relevant areas.

In its previous evaluations of AP1000 DCD, Section 3.8.2, the staff identified acceptance criteria based on the design meeting relevant requirements in 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records"; in GDC 16, "Containment Design"; in GDC 51, "Fracture Prevention of Containment Pressure Boundary"; and in GDC 53, "Provisions for Containment Testing and Inspection." The staff found that the AP1000 containment design was in compliance with these requirements, as referenced in NUREG-0800, Section 3.8.2, and determined that the design of the AP1000 containment, as documented in the AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria. In its previous evaluations of AP1000 DCD Section 3.8.2, the staff also concluded that satisfaction of the relevant requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena"; GDC 4, "Environmental and Dynamic Effects Design Bases"; and GDC 50, "Containment Design Basis," will be demonstrated upon completion of the American Society of Mechanical Engineers (ASME) design report by the COL applicant.

In AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.2 of the certified design:

1. As a result of the extension of the AP1000 design from hard rock sites to sites ranging from soft soils to hard rock, various seismic re-analyses of the Nuclear Island (NI) structures (containment, auxiliary, and shield buildings) were performed. The design of the steel containment structure for seismic loading relies upon the use of the equivalent static method, in which the acceleration profile calculated from the dynamic seismic analysis of a stick model

representation of the steel containment is applied as a static load (mass times acceleration). The dynamic seismic re-analyses of the AP1000 NI, to extend the seismic design basis to soil sites, includes the same stick model representation of the steel containment. In TR-09, the applicant compared the corresponding acceleration profiles obtained from the soil-structure interaction analyses for the various soil sites to the original hard rock acceleration profile used to design the steel containment. On the basis of this comparison, the applicant concluded that the steel containment design is adequate for the range of soil sites considered.

2. The applicant eliminated the COL information item for design of the containment vessel adjacent to large penetrations. The basis for this change is documented in TR-09. The applicant indicated that the applicable changes have been incorporated into the DCD. Therefore, the combined license application (COLA) applicants are no longer required to address this item.

3. Section 3.8.2.7 of DCD Revision 16 was revised to remove the requirement that the in-service inspection of the containment vessel will be performed in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWE, and that this is the responsibility of the COL applicant. This requirement was replaced by the statement that the in-service inspection of the containment vessel will be performed.

4. The applicant undertook efforts, based on feedback from the staff transmitted in an NRC letter dated October 15, 2009, to redesign the shield building. The applicant revised the design of the shield building and submitted the details of this redesign in a separate shield building report which accounts for the revised NI model subjected to seismic and other applicable loads.

The staff has performed a confirmatory seismic analysis of the NI and discovered errors in the Westinghouse model used in the SSI seismic analyses. These errors occurred during the conversion of the [ ] NI20 model to the [ ] NI20 model used in the SSI analyses. The applicant indicated that it would correct the model and rerun the seismic SSI analyses. The new seismic SSI analysis was submitted on March 22, 2010, as APP-GW-S2R-010, Revision 4 (TR-03). The staff finds that both seismic loads (member forces) for structures and the design-basis ISRS have changed at some locations. The applicant's reanalysis results and RAIs, discussed in Sections 3.8.2 through 3.8.5 of the SER and the shield building SER, reflect the results of the reanalysis.

3.8.2.1 Description of the Containment

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that Figure 3.8.2-4, Sheet 6 of 6, which presents a typical containment electrical penetration, has been revised in TR-134, Revision 0. In RAI-SRP3.8.2-SEB1-06, the staff requested the applicant to explain why wedge supports on the outside of containment are used for this penetration. If they provide support to the containment penetration in the vertical and/or horizontal directions, the staff asked how the containment deformation is due to thermal and other loads accommodated or considered in the analysis. The applicant was also requested to address this item for other penetrations where this issue is applicable.

In a letter dated February 19, 2009, the applicant stated that in Figure 3.8.2-4 of the AP1000 DCD, Revision 17, the typical containment electrical penetration design was replaced with a design that does not include wedge supports at the shield building end. AP1000 DCD, Revision 17, Sections 3.8.2.1.6 and 3.8.2.4.2.5, also include revisions to information on the electrical penetrations. The staff reviewed the AP1000 DCD, Revision 17 and verified that Figure 3.8.2-4 for the typical containment electrical penetration design does not include wedge supports, and, thus, eliminates an undue constraint on the penetration. Therefore, the staff finds that RAI-SRP3.8.2-SEB1-06 is resolved.

3.8.2.2 Applicable Codes, Standards, and Specifications

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that Section 3.8.2.2, as well as other sections of the DCD related to structures, refer to AP1000 DCD Section 1.9 for discussion of conformance with RGs. The staff finds that for RG 1.7," Control of Combustible Gas Concentrations in Containment," and RG 1.57," Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," the AP1000 DCD is in accordance with earlier revisions of the RGs. The AP1000 DCD indicates that RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," is not applicable to the AP1000 DC and that Section 17.5 of the AP1000 DCD defines the responsibility for a plant maintenance program. RG 1.199, "Anchoring Components and Structural Supports in Concrete," which is identified as another applicable guide in NUREG-0800 Section 3.8, is not described at all in Section 1.9 of the AP1000 DCD.

In RAI-SRP3.8.2-SEB1-02, the staff requested the applicant to indicate whether the design, construction, and inspection of the AP1000 plant are in accordance with the current RGs and whether RG 1.199, "Anchoring Components and Structural Supports in Concrete," was used to meet the NRC's regulatory guidance for the design, evaluation, and quality assurance (QA) of anchors (steel embedments).

In a letter dated April 17, 2009, the applicant provided its response to this RAI. The staff's assessment of the response for each RG is discussed below:

RG 1.7

The applicant's response indicated that the current AP1000 certified design is consistent with Revision 3 of RG 1.7 (issued in March 2007). The AP1000 containment design is a passive system, using convective mixing. Design features promote free circulation of the containment atmosphere. NUREG-1793 documents an analysis of the effectiveness of the passive mixing.

The staff found that the applicant did not discuss whether the hydrogen generated loads were evaluated in accordance with RG 1.7 for the containment acceptance criteria and RG 1.57 for the applicable load combinations.

RG 1.57

The applicant's response indicated that RG 1.57, Revision 1 (issued in March 2007) endorses ASME Boiler and Pressure Vessel Code (B&PV), Section III, "Rules for Construction of Nuclear

Facility Components," Division 1, Subsection NE, "Class MC Components," 2001 Edition with 2003 Addenda and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition with 2003 Addenda.

The applicant's response also indicated that the CV is designed to meet the requirements of ASME B&PV Code, Section III, 2001 Edition including the 2002 Addenda. The 2003 Addenda did not include any requirements that impact the design of the CV described in the AP1000 DCD. There are only two changes (which are in Subsection NE-5000, "Examination") and they are related to the examination of the welds and do not impact the design. Therefore, the applicant concluded that the CV design is in conformance with this RG.

Since the response did not discuss the regulatory positions in RG 1.57, the applicant was requested to specifically confirm whether all of the regulatory positions presented in RG 1.57, Revision 1, have been satisfied for the AP1000 plant.

RG 1.199

The response indicated that RG 1.199, "Anchoring Components and Structural Supports in Concrete," (Revision 0), was issued in November 2003, to provide guidance to licensees and applicants on methods acceptable to the NRC staff for complying with the NRC's regulations in the design, evaluation, and QA of anchors (steel embedments) used for component and structural supports on concrete structures. As a result of studies and tests performed, questions were raised regarding the design methodology used in Appendix B to ACI-349-80. After an extensive review of available test data, the ACI-349 Code committee issued a revision to ACI-349, Appendix B in February 2001.

RG 1.199 generally endorses Appendix B to ACI-349-01, with exceptions in the area of load combinations.

- The AP1000 NI concrete structures are designed to meet the requirements of the ACI-349-01 Code, including Appendix B on the design of anchors in concrete.

- Following the release of this RG, the load combinations used in the design of NI concrete structures were reviewed and approved by the NRC in the AP1000 DC for the HR sites.

The attached table to the RAI response provided itemized conformance with the regulatory positions of this RG.

In the RAI response above, the applicant did not provide any information on the provisions in RG 1.160 (10 CFR 50.65, "Maintenance Rule").

In the audit conducted during the week of May 4, 2009, the staff discussed with the applicant all the missing information associated with the above key RGs. In a letter dated September 29, 2009, the applicant transmitted a revised RAI response, which provided additional information. The staff reviewed the response and determined that it did not fully address all of the concerns related to the RGs. Therefore, the applicant was requested to address the following remaining items:

1. Explain whether the regulatory positions in RG 1.7, Revision 3 and RG 1.57, Revision 1, related to containment structural integrity under the hydrogen generated pressure loads, were satisfied or provide justification for the use of alternate methods.

2. Explain whether the regulatory positions in RG 1.57, Revision 1, related to the design limits and load combinations, were met.

3. Document in the AP1000 DCD the testing and inservice surveillance programs for plant structures. Monitoring and maintenance criteria are identified in NUREG-0800 Sections 3.8.1 through 3.8.5. With the exception of containments, each of these sections identifies that RG 1.160 is applicable. Therefore, confirm that RG 1.160 is applicable for the maintenance of structures at the plant and confirm that it will be followed when implementing 10 CFR 50.65. Also, revise the AP1000 DCD to reflect the applicability of RG 1.160, Revision 2. The performance of inservice inspection of containment is required by 10 CFR 50.55a and ASME B&PV Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components.

4. Revise the AP1000 DCD to indicate that RG 1.199 (2003) is applicable for anchoring components and structural supports in concrete for the AP1000 plant.

In response to the above requests, the applicant's letters dated July 2, and August 25, 2010, indicate that the AP1000 CV design is consistent with the guidance of RG 1.7, Revision 3, and RG 1.57, Revision 1. Details of the methods used to address the hydrogen generated loads, load combinations, and design limits for containment design are presented in the response to RAI-SRP3.8.2-SEB1-03. Since the design of the CV is consistent with these two RGs, the staff finds that Items 1 and 2 identified above have been adequately addressed.

To address the inservice inspection of plant structures, the applicant proposed to revise the text in AP1000 DCD Sections 3.8.3, 3.8.4, 3.8.5 and 3.8.6, and in AP1000 DCD Tables 1.8-2 and 1.9-1, to indicate that the COL applicant is responsible for establishing a structures inspection program consistent with the maintenance rule in 10 CFR 50.65 and the guidance provided in RG 1.160. This addresses the inservice testing, inspection, or special maintenance requirements for the seismic Category I and seismic Category II structures. Since the AP1000 DCD will be revised to identify the requirements for the COL applicants to develop the inservice inspection and maintenance program for structures, the staff concludes that Item 3 has been adequately addressed.

To address Item 4, the applicant proposed to revise the text in AP1000 DCD Sections 3.8.3, 3.8.4 and 3.8.5, and in AP1000 DCD Table 1.9-1, to indicate that the design of anchorage to concrete is in accordance with ACI-349-01, Appendix B, and is in conformance with RG 1.199, Revision 0. Since the AP1000 DCD will be revised to require that concrete anchors will be designed in accordance with RG 1.199, Revision 0, the staff concludes that Item 4 has been adequately addressed. The staff's evaluation of the inservice inspection requirements for containment is discussed later in Section 3.8.2.6 of this SER.

Based on the above discussion, the AP1000 DCD revision to incorporate the mark-ups proposed in the RAI response will be tracked as Confirmatory Item CI-SRP3.8.2-SEB1-02.

3.8.2.3 Loads and Load Combinations

During the review of AP1000 DCD Tier 2, Revision 16, the staff identified in RAI-SRP3.8.2-SEB1-03 a concern that Table 3.8.2-1 does not include several load combinations that are applicable to the CV design. These missing load combinations are described in 10 CFR 50.44; RG 1.57; RG 1.7; and NUREG-0800 Section 3.8.2.II, Acceptance Criteria 3.B.iii. In a letter dated February 19, 2009, the applicant provided a response to this RAI. The response provided the technical basis for not considering the load combination for post flooding condition and also explained how the loading combination for external pressure due to inadvertent actuation of the fan coolers was considered. Further, the load combination with operating-basis earthquake (OBE) for fatigue consideration was not required because the conditions specified in the ASME B&PV Code, Section III, Division 1, Subsection NE were satisfied. However, the staff determined that insufficient information was provided to explain the remaining missing load combinations and the external pressure loading imposed on the containment.

In a letter dated February 17, 2010, the applicant provided a revised response to address the remaining questions on the missing load combinations and the question on the correct external pressure to be used for the containment design. Based on the staff's review of this RAI response and the related response to RAI-TR09-08, Revision 4, the staff determined that several items still needed to be addressed. Therefore, in a follow-up RAI, the staff requested that the applicant explain why the load combinations that combine wind load with design pressure load and combine tornado wind load plus external pressure load do not appear in the proposed revision of AP1000 DCD Table 3.8.2-1. Also, the AP1000 DCD table should identify the values for the different pressures and the corresponding temperatures inside and outside containment that are used in each of these load combinations. In addition, the applicant was requested to clarify the response given regarding the hydrogen generated load evaluations for containment. These clarifications are needed to ensure that the applicable loads and load combinations described in 10 CFR 50.44; RGs 1.57 and 1.7; and NUREG-0800 Section 3.8.2, were considered.

In response to the above requests, the applicant's letters dated July 2, and August 25, 2010, indicate that the design wind load is small, within the operating pressure of the containment, which ranges from 100.0 to 108.3 kilopascals (kPa) (-0.2 to 1.0 pounds per square inch gauge (psig)). This occurs because the shield building, which surrounds the containment, has limited openings in the vent area at the top of the cylindrical shield building wall. Therefore, the load combination, that combines design wind load plus internal design pressure of 508.2 kPa (59 psig) is not included in Table 3.8.2-1. For the load combination of tornado wind load plus external pressure, the RAI response indicates that the effects of the tornado wind load for the AP1000 containment reduces the external pressure. Therefore, there is no need to consider this load combination. The staff finds that the RAI response for these two load combinations is acceptable because the effect of the wind load is considered to be negligible and the tornado load reduces the effect of the containment external pressure load.

For the definitions of the different pressures and corresponding temperatures inside and outside containment that are used in the load combinations presented in AP1000 DCD Table 3.8.2-1, the RAI response indicates that they are presented in the response to RAI-TR09-08, Revision 5.

The staff confirmed that the four different pressures and temperatures are defined in the response to RAI-TR09-08. The adequacy of these pressure and temperatures is evaluated separately under the staff's assessment of RAI-TR09-08.

The RAI response provided clarifications and also proposed to make revisions in the AP1000 DCD to explain how the hydrogen generated pressure and hydrogen burn loadings were considered in accordance with 10 CFR 50.44. In addition, as noted in the staff's evaluation of RAI-SRP3.8.2-SEB1-02 above, the design of the AP1000 CV for hydrogen generated loadings is consistent with the guidance of RG 1.7, Revision 3, and RG 1.57, Revision 1. The staff finds that the information provided and the proposed changes to the AP1000 DCD are acceptable because the design is performed in accordance with 10 CFR 50.44, applicable RGs, and is consistent with NUREG-0800 Section 3.8.2.

Based on the above discussion, the AP1000 DCD revision to incorporate the mark-ups proposed in the RAI response will be tracked as confirmatory item CI-SRP3.8.2-SEB1-03.

3.8.2.4 Design and Analysis Procedures

During the review of AP1000 DCD Tier 2, Revision 16, the staff identified that Section 3.8.2.4.1.2, which describes the local analyses for the penetrations of the steel containment, has been revised from its previous revision. The revision relies on the use of a new 3D finite element model of the entire containment, which includes the penetrations rather than using separate localized models of the penetrations. In RAI-SRP3.8.2-SEB1-04, the staff requested that the applicant provide a more detailed explanation of: (1) the new 3D finite element model of the entire containment described in Section 3.8.2.4.1.2 used for the local evaluation near penetrations; and (2) the axisymmetric model described in Section 3.8.2.4.1.1 and Appendix 3G, which is used for the analysis of the containment in regions away from penetrations. This information is needed to ensure that the revised model of the entire containment, developed for local analysis of penetrations, is adequate to capture the containment response.

In a letter dated April 29, 2009, and in a subsequent letter dated July 7, 2009, the applicant provided information to address this RAI. The staff reviewed this response and concluded that the applicant has provided a description of the 3D finite element model of the entire containment, and a description of the finite element model of the containment used for the local evaluation near large penetrations. The response indicated that more detailed information is presented in TR-09. The staff's evaluation of TR-09 is presented below. The staff reviewed the RAI response and concluded that the analysis approach is consistent with industry methods and guidance presented in NUREG-0800 Sections 3.7 and 3.8. In the July 7, 2009, RAI response, the applicant proposed several changes to be included in a future revision of the AP1000 DCD. Therefore, the AP1000 DCD revision to incorporate the mark-ups proposed in the RAI response will be tracked as Confirmatory Item CI-SRP3.8.2-SEB1-04.

Containment Penetrations Technical Report TR-09

The applicant submitted APP-GW-GLR-005, TR-09, "Containment Vessel Design Adjacent to Large Penetrations," (current version is Revision 3, June 2009) to summarize the design of CV reinforcement adjacent to large penetrations. The design of the penetrations in the TR-09

report also considers the results of the seismic evaluations conducted to extend the applicability of the AP1000 CV design to soil sites.

The applicant completed the design and analyses of the CV reinforcement for the large penetrations (two equipment hatches and two airlocks), and submitted the evaluation to the NRC as Westinghouse Report APP-GW-GLR-005, Revision 0 (TR-09, Revision 0) in May 2006. However, the main steam and feedwater penetrations were not addressed in TR-09, Revision 0. In RAI-TR09-01, the staff requested the applicant to include the design and analysis details for the main steam and feedwater penetrations in TR-09.

In a letter dated September 5, 2007, the applicant indicated that Section 2.6 had been added to Revision 1 of TR-09, describing the design of the main steam and feedwater penetration reinforcement, and that the penetration assemblies are connected to the vessel by expansion bellows, thus preventing significant cyclic thermal and mechanical loading in the SCV.

Subsequently, during the October 2007 audit, the applicant provided report number APP-MV50-S2C-012, Revision 2, "Design of Containment Vessel Penetration Reinforcement," which included the detailed design calculations for the main steam and feedwater penetration reinforcement. The staff later reviewed this report and found that it adequately described the design of penetration reinforcement for the main steam, feedwater, and the start-up feedwater penetrations. During the October 2007 audit, the staff raised a concern that TR-09, Revision 1, did not address the fuel transfer tube penetration. The staff requested the applicant to provide information related to the design of the fuel transfer tube penetration comparable to the level of detail provided for the main steam and feedwater penetrations.

In a letter dated June 4, 2009, the applicant transmitted TR-09, Revision 3, which included the additional section on the design of containment penetration reinforcement for other penetrations, including the fuel transfer tube penetration. The staff reviewed TR-09, Revision 3 and concluded that sufficient information was provided to describe the design procedure for the other mechanical and electrical penetration reinforcements. The staff noted that the design procedure is consistent with accepted analytical methods for design of containment penetration reinforcements and is in accordance with the provisions of the ASME B&PV Code, Section III, Subsection NE, for metal containments.

On the basis that the applicant completed and documented the design of the major containment penetrations and documented the design procedure for the other containment penetrations, in accordance with the provisions of the ASME B&PV Code, Section III, Subsection NE, for metal containments, the staff considers RAI-TR09-01 resolved.

In TR-09, Revision 0, the applicant attempted to justify the use of seismic loading derived from the initial hard-rock site condition for the design/analysis of containment penetrations for soil sites. However, the information provided was insufficient for the staff to conduct its review for the extension of the evaluation for soil sites. Therefore, in RAI-TR09-02, the staff requested that the applicant provide the necessary quantitative information in TR-09 to specifically demonstrate the design adequacy of containment penetrations for all soil conditions.

In its response dated September 5, 2007, the applicant indicated that with the exception of the large penetrations (equipment hatches and personnel airlocks), the CV design was completed

for the HR site condition and was reviewed by the NRC during the HR DC, and that this design has not changed. The applicant referenced comparisons included in TR-09, Revision 1, demonstrating that the HR design forces are still applicable. The staff reviewed Figure 2-10 of TR-09, Revision 1, which compares member force and moment results from the dynamic analyses for all soil cases, to the certified HR design member forces and moments. The HR design values envelop the corresponding values for all soil sites. On this basis, the staff concluded that the overall design of the CV, based on the HR site, is also acceptable for the range of soil sites evaluated by the applicant. Therefore, RAI-TR09-02 is resolved.

Since design details for the penetrations included in TR-09, Revision 0, were not provided, the staff requested in RAI-TR09-03 that the applicant include appropriate design information (geometry, material and material properties, dimensions and wall thicknesses) for each penetration in TR-09, and specify the ASME B&PV Code, Class MC jurisdictional boundaries for each penetration.

In a letter dated September 5, 2007, the applicant indicated that typical design information for the penetrations is provided in the AP1000 DCD. This material has now been included in Appendix A of the TR-09 report. Penetration assemblies, such as those shown in the upper figure on AP1000 DCD Figure 3.8.2-4 (Sheet 4 of 6), are ASME B&PV Code Class 2. Expansion bellows and guard pipes are ASME B&PV Code Class 2 or Class MC. The penetration assemblies are welded to sleeves that are ASME B&PV Code Class MC. Process piping welded directly to the vessel, such as shown in the lower figure in AP1000 DCD Figure 3.8.2-4 (Sheet 4 of 6), is ASME B&PV Code Class 2.

The material of construction is SA738 Grade B for the vessel shell, insert plates and nozzle necks of penetrations with inside diameters greater than 24". For penetrations less than 24" inside diameter and greater than 2" nominal diameter, forgings of SA350 LF2 material are used for the nozzle neck.

Other design requirements for the mechanical penetrations, as stated in the applicant's letter dated September 5, 2007, are as follows:

- Design and construction of the process piping follow the ASME B&PV Code, Section III, Subsection NC. Design and construction of the remaining portions follow the ASME B&PV Code, Section III, Subsection NE. The boundary of jurisdiction is according to the ASME B&PV Code, Section III, Subsection NE.

- Penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation.

- Guard pipe assemblies for high-energy piping in the containment annulus region between the containment shell and shield building that are part of the containment boundary are designed according to the rules of Class MC, Subsection NE, of the ASME B&PV Code.

- Bellows are stainless steel or nickel alloy and are designed to accommodate axial and lateral displacements between the piping and the CV. These displacements include thermal growth of the main steam and feedwater piping during plant operation, relative

seismic movements, and containment accident and testing conditions. Cover plates are provided to protect the bellows from foreign objects during construction and operation. These cover plates are removable to permit inservice inspection.

The staff finds that the applicant provided design details sufficient to enable the staff to proceed with its review of the penetrations; therefore, RAI-TR09-03 is resolved.

Based on the review of TR-09, Revision 0, the staff noted that there was insufficient description of the load cases analyzed. Therefore, in RAI-TR09-05, the staff requested that the applicant describe the loads analyzed and how they were combined, and whether the containment post-loss-of-coolant accident (LOCA) flooding load was included in the load combinations.

In a letter dated September 5, 2007, the applicant indicated that Section 2.3 of TR-09 had been revised to describe the individual loads and their combinations; and that the post-LOCA flooding event is not considered in the load combination because it is enveloped by other design load combinations. During the October 2007 audit, the staff found that the load combinations in the AP1000 DCD and in the CB&I Containment Vessel Design Report (APP-MV50-S3R-003) are the same, but the load combinations listed in TR-09 are different. The staff requested that the applicant explain the differences or demonstrate that they are all consistent.

The issue of adequacy of the containment load combinations is also addressed under RAI-SRP3.8.2-SEB1-03, which is evaluated in Section 3.8.2.3 of this SER. Based on that evaluation, RAI-SRP3.8.2-SEB1-03 is classified as Confirmatory Item CI-SRP3.8.2-SEB1-03, pending revision of the AP1000 DCD to incorporate the mark-ups proposed in the RAI response. In addition, the applicant's letter dated June 18, 2010, indicated that TR-09 will be revised to be consistent with the load combinations in the proposed revisions to AP1000 DCD Table 3.8.2-1. Accordingly, the staff has identified the status of RAI-TR09-05 as Confirmatory Item CI-RAI-TR09-05.

There were no results presented in TR-09, Revision 0, for buckling analyses of the containment. Therefore, in RAI-TR09-07, the staff requested that the applicant include in TR-09, Revision 0, a detailed description of buckling analysis and results.

In a letter dated September 5, 2007, the applicant indicated that Section 2.4.2.2 had been added to TR-09, Revision 1, to provide the requested information. During the May 19-23, 2008 audit, the staff reviewed calculation APP-MV50-S2C-010, Revision 0, "3D Model - Analysis of Large Penetrations," and concluded that the buckling analyses were appropriately considered and that the calculated stresses were less than the acceptance limits. Therefore, RAI-TR09-07 is resolved.

The staff noted that AP1000 DCD, Revision 15, as well as AP1000 DCD, Revisions 16 and 17, indicate that the design external pressure is 2.9 pounds per square inch differential (psid). However, in TR-09, the applicant presented a justification for reducing the design external pressure from 2.9 psid to 0.9 psid, and stated that an estimate of the external pressure was provided in the response to DSER OI 3.8.2.1-1. Therefore, in RAI-TR09-08, the staff requested that the applicant demonstrate the design adequacy of the containment penetrations and the steel CV for a design external pressure of 2.9 psid.

In its Revision 2 response to RAI-SRP6.2.1.1-SPCV-07, dated December 14, 2009, the applicant stated that the design external pressure of 2.9 psid is used in the design load combination and the lower external pressure of 0.9 psid is a more credible external pressure used to define Service Level A and D load combinations. Because the Service Level A load combinations include thermal loads, the applicant evaluated different events at various external temperature conditions to demonstrate that 0.9 psid bounds the external pressure excursions that could occur on a cold day.

In a letter dated February 17, 2010, the applicant provided information to address questions raised regarding the temperature and external pressure loads used for design of the containment. The staff's review of this information determined that additional information was required. In a follow-up to RAI-TR09-08, the staff requested that the applicant provide the following:

a. In Table 1 of the RAI response, the results show a trend of higher external pressure as the outside temperature increases. However, the analysis is limited to ≤ -7.2°Celsius (C) (19°Fahrenheit (F)), for which the external pressure is 0.98 pounds per square inch (psi). Provide the technical basis for limiting the analysis to -7.2°C (19°F) for the outside temperature.

b. After reviewing the RAI response and the proposed revision to AP1000 DCD Table 3.8.2-1, it is not clear what temperature gradient/external pressure combination is used in the Service Level A load combination notated by Footnotes 3 and 5. Describe in detail, the pressure and temperature condition used in this Service Level A load combination, and the technical basis for concluding it is the worst case. Include this information in AP1000 DCD Section 3.8.2 and in TR-09. Revise AP1000 DCD Table 3.8.2-1 footnotes to reference AP1000 DCD Section 3.8.2 that describes this loading condition.

c. The staff noted a number of inconsistencies between proposed AP1000 DCD Table 3.8.2-1 and the latest TR-09 Table 2-4, both of which identify the applicable load combinations for design of the containment structure. Revise these tables so that they are consistent, or provide the technical basis for the inconsistencies.

d. The maximum external pressure is no longer listed as 0.9 psi in the proposed revision to AP1000 DCD Table 3.8.2-1. For consistency, ensure that all references to the 0.9 psi external pressure in both the AP1000 DCD and TR-09 are appropriately revised.

Based on the applicant's letter dated July 30, 2010, much of the transient information provided previously was revised because a containment vacuum relief system was added with an actuation point of 0.8 psid. Based on the external pressure that the containment vacuum relief system can mitigate, a conservative external design pressure is defined as 1.7 psid. This design external pressure is combined with a coincident temperature of -40°C (-40°F) outside air temperature, which corresponds to -28°C (-18.5°F) for the CV shell region that is not insulated and 21.1°C (70°F) for the shell region that is insulated from the cold outside air. Additional information on the appropriate temperatures for this external pressure loading condition is discussed under RAI-SRP3.8.2-CIB1-01 in Section 3.8.2.5 of this SER. The applicant's July 30, 2010, letter provided the proposed changes to AP1000 DCD Section 3.8.2 related to

the revised pressures and temperatures for design of the containment. The letter also indicated that TR-09 will be revised to be consistent with the AP1000 DCD changes. The staff's review of the letter concluded that the information provided in the response described the various pressure and temperature loadings to be used for design of the containment, and thus, addressed all of the staff's prior concerns for defining the pressure and temperature loads on the containment. Therefore, this is Confirmatory Item CI-RAI-TR09-08, pending revision of the AP1000 DCD to incorporate the proposed changes in the RAI response and revision of TR-09 to be consistent with the information in the AP1000 DCD.

3.8.2.5 Materials, Quality Control, and Special Construction Techniques

In Revision 16 to the AP1000 DCD, the applicant proposed changes to the supplementary requirements of the CV shell material specification. This resulted in changes to the AP1000 DCD in Section 3.8.2.6. In a letter dated May 11, 2007, Westinghouse Electric Company, LLC (the applicant) submitted AP1000 Standard COL TR-113, "AP1000 Containment Vessel Shell Material Specification," APP-GW-GLN-113, Revision 0 to provide the technical justification for the proposed changes.

Revision 15 to the AP1000 DCD, Section 3.8.2.6 specified the basic containment vessel (CV) material as SA-738, Grade B plate. The procurement specification for this plate material is required to include supplemental requirements S17, "Vacuum Carbon-Deoxidized Steel" and S20 "Maximum Carbon Equivalent for Weldability." The applicant has investigated the availability of SA-738, Grade B plate material (with S17 supplementary requirement) in the United States as well as in all the large, steel-producing countries in the world. The investigation determined that steel producing mills do not use an S17 process, but, rather, use a supplementary requirement S1 process to get similar high-quality, vacuum-degassed steel.

The applicant proposed to correct the AP1000 DCD in Revision 16 to specify supplementary requirement S1 instead of the currently specified supplementary requirement S17. The applicant provided the following technical justification in support of the proposed change to AP1000 DCD Section 3.8.2.6.

The use of a vacuum carbon-deoxidized (VCD) process in steel production typically applies to certain grades of chromium-molybdenum (Cr-Mo) steels where carbon contents are lower and reduced silicon content is beneficial. The VCD process allows oxygen and carbon to react in the molten steel and evolve as carbon monoxide, which is drawn off by the vacuum. While under vacuum, other gases, such as hydrogen and nitrogen, also tend to be removed from the steel. Reducing the oxygen content by VCD reduces the need for the addition of other deoxidizing additions such as silicon or aluminum. Steels treated by VCD have a specified silicon content of 0.12 percent maximum that is lower than the normally specified range of silicon content. This process is beneficial in Cr-Mo steels that are susceptible to temper embrittlement during elevated-temperature service. Silicon is one of the impurity elements that contribute to the loss of toughness. By reducing the silicon content of the steel the tendency for temper embrittlement is reduced. The use of the VCD process for vacuum degassing of SA-738 plate material was discussed with a metallurgist from a large, domestic-steel plate producer. The steel producers in the United States typically do not use VCD for plate materials like SA-738. For this reason, requiring supplementary requirement S17 to be used for the production of SA-738 plate material is somewhat of an anomaly. Therefore, the supplementary

requirement S1, "Vacuum Treatment," is more appropriate for this type of material because S1 requires the steel to be made by a process, which includes vacuum degassing while molten by a suitable practice selected by the steel manufacturer or purchaser.

In addition, Revision 16 to the AP1000 DCD, Section 3.8.2.6 was changed to specify the lowest service temperature of -28°C (-18.5°F) instead of -26.1°C (-15°F), which was previously stated in Revision 15 of the AP1000 DCD. TR-113 did not specify the change to the service temperature nor provide any justification for this change in service temperature as required by 10 CFR 52.63(a)(1).

The staff reviewed the applicant's request to revise AP1000 DCD, Section 3.8.2.6 concerning the supplementary requirements of the CV shell material specification and found it acceptable because of following reasons.

The SA-738, Grade B plate material was approved for use in construction of metal CVs in ASME Code Case N-655, Section III, in February 2002. This plate material was also incorporated into Table 1A of Section II, Part D in the 2002 Addenda to the 2001 Edition of the ASME B&PV Code. The NRC conditionally accepted ASME Code Case N-655 in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Revision 33 in August 2005. The conditions that the NRC placed on the use of SA-738 plate material were to specify the use of supplementary requirements S17 and S20 when using SA-738 material for construction of CVs. The two conditions were needed to ensure adequate material properties and weldability of the CV material. The ASME Code, Section III, exempts SA-738, Grade B, material up to 4.4 centimeters (cm) (1.75 inch (in)) of thickness from post-weld, stress-relief heat treatment. Because the welds in CV material thickness up to 4.4 cm (1.75 in) thick will not be stress-relieved, higher residual stresses will be present in the welds. Also, the material will likely be procured in the quenched and tempered condition. Welding will reduce the impact properties of the material in the heat affected zone. Requiring the use of vacuum degassed steel will ensure adequate material properties because nonmetallic inclusions, such as oxides and silicates will be minimized as a result of the vacuum degassing of the steel. S17 supplementary requirement was specified to accomplish the vacuum degassing of the steel. Requiring supplementary requirement S20 and a carbon equivalent weldability check will ensure that the steel is readily weldable.

The staff specified the use of S17 for SA-738 material because at the time of the review of ASME Code Case N-655, S17 was the only requirement clearly listed in the specification that would provide for vacuum degassing of steel. Supplementary requirement S1 was also available for SA-738 plate material; however, S1 is listed in SA-20, "General Requirements for Steel Plates for Pressure Vessels," which is referenced in the SA-738 specification. Therefore, in order to impose the S1 requirement in the CV, the designer would have to specify two specifications instead of one. The purpose of the staff's condition was to specify the use of vacuum degassed steel. Imposing an S1 supplementary requirement would accomplish this goal. Furthermore, at the time of approval of ASME Code Case N-655 neither the staff nor the applicant was aware that the steel producers had limited S17 to the production of Cr-Mo steels. Since the discovery of this situation, the ASME Code has approved a revision to the ASME Code Case N-655-1, which correctly specifies the use of S1 and S20 supplementary requirements for the use of SA-738 plate material. On this basis, the staff concludes that the

proposed revision to AP1000 DCD, Section 3.8.2.6 to specify supplementary requirement S1 meets the requirements of 10 CFR 50.55a and the ASME Code, Section III, and is acceptable.

In regard to the service temperature of the CV, Tier 2, Section 3.8.2.6 of the AP1000 DCD, describes the materials used to fabricate the CV. The material selected satisfies the lowest service metal temperature requirement, established by analysis for the portion of the vessel exposed to the environment when the ambient air temperature is -40°C (-40°F). APP-GW-GLN-113 (TR-113), "AP1000 Containment Vessel Shell Material Specification," Revision 0, submitted by the applicant in a letter dated May 11, 2007, also revised this section to specify the lowest service temperature of -28.1°C (-18.5°F) instead of -26.1°C (-15°F), which was previously stated in Revision 15 of the AP1000 DCD. TR-113 did not specify the change to the service temperature nor provide any justification for this change in service temperature as required by 10 CFR 52.63(a)(1). In NUREG-1793, Section 3.8.2.6, the NRC staff approved 9.4°C (15°F) as the lowest service temperature based on the staff's review of the applicant's calculation APP-PCS-M3C-002, Revision 1, "AP1000 Containment Shell Minimum Service Temperature." Therefore, the staff requested that the applicant provide its reason and justification for the change in minimum service temperature of the CV in accordance with 10 CFR 52.63(a)(1), along with the analysis that supports the new service temperature proposed in Revision 16 of the AP1000 DCD. This was previously addressed in RAI-SRP3.8.2-CIB1-01.

In a letter dated July 22, 2008, the applicant stated that an additional scenario was postulated for the CV shell analysis, which determined that the CV will be subjected to a service metal temperature of -28.1°C (-18.5°F). This evaluation postulated that an SSE event occurred in conjunction with -40°C (-40°F) outside temperature and inadvertent actuation of active containment cooling. APP-GW-GLR-005 (TR-09) only described the analysis, and inadvertently did not include the corresponding service metal temperature.

Since TR-09 did not include the analysis or the service metal temperature, the NRC staff could not confirm that -28.1°C (-18.5°F) was the lowest service metal temperature of the CV shell, which is fabricated from SA-738 Grade B material. This material must meet the requirements of NE-2000 for fracture toughness (Charpy V-notch test) in the as-welded condition for thicknesses up to and including 4.4 cm (1.75 in), and in the post-weld heat treated condition for thicknesses greater than 4.4 cm (1.75 in). The minimum service temperature is used to determine the testing temperature for the Charpy V-notch tests required by the ASME Code, Section III, Subsections NE 2300 and NE-4300. Previously, the applicant stated in its letter dated April 22, 2003, that the SA-738, Grade B plate material will be procured using the service metal temperature of -26.1°C (-15°F) (i.e., -48.3°C (-55°F) Charpy V-notch test temperature as required by the ASME Code, Section III, Subsections NE-4335.2(b)(2) and Tables NE-4622.7(b)-1, note (2)(b)(1)) in order to account for degradation during welding of the heat affected zone in the base material. In addition, the applicant stated in a letter dated March 13, 2003, that the previous analysis added a -13.3°C (8°F) conservative factor to obtain a minimum service metal temperature of -26.1°C (-15°F).

Therefore, the NRC staff required additional information to verify the minimum service metal temperature including the details of the analysis (e.g., calculation methodology, assumptions made, similarities/differences from previous analysis, etc.) to confirm that -28.1°C (-18.5°F) is the lowest service metal temperature to ensure that the material will be tested to have adequate

toughness for the design and environment the containment shell will experience. The staff also requested clarification of whether the conservative factors described in the applicant's letter dated March 13, 2003, were used in this analysis or provide justification for not including these conservative factors.

In a letter dated May 7, 2009, the applicant stated that the additional information was provided in APP-MV50-Z0C-020, Revision 0. However, the NRC staff requested that the assumptions made along with the similarities/differences from the previous analysis (for Revision 15 of the AP1000 DCD) be addressed. In response to Revision 2 of RAI-SRP3.8.2-CIB1-01, the applicant provided in a letter dated September 17, 2009, the assumptions and differences between the analyses. The applicant stated that the original analysis for -26.1°C (-15°F) minimum service metal temperature in Revision 15 of the AP1000 DCD was performed by a hand calculation using a simple radial heat balance model, and then added an -13.3°C (8°F) conservatism factor. The minimum service metal temperature of -28.1°C (-18.5°F) was determined by a WGOTHIC computer code, using a free/forced convection model. This model calculated a higher heat transfer coefficient; thereby, resulting in a lower minimum service metal temperature (-28.1°C (-18.5°F) versus -26.1°C (-15°F)). The NRC staff notes that WGOTHIC is currently used in other pressure and temperature determinations for operating reactors. In addition, WGOTHIC has its own inherent conservatisms within the computer code. Therefore, the NRC staff determined that the use of WGOTHIC computer code is valid in determining the minimum service metal temperature for the steel containment.

In a letter dated February 17, 2010, the applicant performed a new WGOTHIC analysis documented in APP-MV50-Z0C-039, Revision 0, which used an outside temperature at -40°C (-40°F) and -34.4°C (-30°F). However, the NRC staff notes that this analysis was not a bounding case, since it used different assumptions for the wind speeds at these two temperatures based on Duluth, Minnesota, meteorological data. The Duluth data documented the wind speed at -34.4°C (-30°F) to be faster than at -40°C (-40°F). Using these temperatures and wind speeds, the -34.4°C (-30°F) case resulted in a higher velocity through the annulus between the containment and air baffle, and thereby, a greater heat transfer coefficient. Therefore, based on the Duluth, Minnesota, weather records, the applicant's analysis determined that the -34.4°C (-30°F) outside temperature condition resulted in minimum service metal temperature of -8.1°C (-0.61°F) versus a minimum service metal temperature of -13.8°C (7.18°F) for an outside temperature of -40°C (-40°F). Since the analysis in APP-MV50-Z0C-039, Revision 0 was not a bounding case, the NRC staff requested that a bounding analysis be performed using an outside temperature of -40°C (-40°F) and a maximum wind speed of 77.24 kilometers per hour (kmph) (48 miles per hour (mph)), used in previous calculations, or provide justification for the validity of the Duluth temperature/wind speed data along with a sensitivity study.

In a letter dated May 10, 2010, the applicant provided an analysis for the loss of alternating current (ac) power (LOAC) transient using an outside temperature of -40°C (-40°F) with a corresponding wind speed of 48 mph, which produced a minimum service metal temperature of -27.2°C (-16.91°F), which is bounded by the -28.1°C (-18.5°F) minimum service metal temperature in the AP1000 DCD. The staff notes that the -8.4°C (16.91°F) temperature included a factor to compensate for any temperature uncertainty in the calculation near the air baffle plate. The bounding case used the LOAC transient in Case 11 of APP-MV50-ZOC-039, Revision 0, by adjusting the wind speed to 77.24 kmph (48 mph). Based on the June 18, 2010,

letter, the applicant stated that the LOAC transient was the limiting event since the inadvertent activation of the containment fan cooler event is no longer credible because the fan coolers are operational. Therefore, the NRC staff considers this to be a bounding condition in determining the minimum service metal temperature and that the -28.1°C (-18.5°F) temperature in the AP1000 DCD is supported by an appropriate analysis. The NRC staff notes that in the letter dated May 10, 2010, the applicant provided a bounding calculation in lieu of justifying the current data in APP-MV50-ZOC-039, Revision 0. However, the applicant did not revise APP-MV50-ZOC-039, Revision 0, to reflect this bounding calculation, and assumes that the results depicted in APP-MV50-ZOC-039, Revision 0, are the result of record for the AP1000 DCD. The NRC staff requests that the applicant revise APP-MV50-ZOC-039, Revision 0, to reference this bounding calculation, since the bounding case was provided in lieu of justifying the current data in APP-MV50-ZOC-039, Revision 0. The NRC staff identifies this as Open Item OI-SRP3.8.2-CIB1-01.

In a letter dated July 9, 2010, the applicant stated that the bounding case provided in the letter dated May 10, 2010, would be incorporated into APP-MV50-ZOC-039. In addition, the applicant stated in letters dated July 30, 2010, and August 16, 2010, that the addition of a vacuum relief system does not invalidate APP-MV50-ZOC-039 for the determination of the minimum service metal temperature. The NRC staff agrees that the bounding calculation for the minimum service metal temperature in APP-MV50-ZOC-039, as modified by letter dated July 9, 2010, is still applicable, since it is calculates the lowest possible service metal temperature corresponding with an outside temperature of -40°C (-40°F). This resolves Open Item OI-SRP3.8.2-CIB1-01.

However, the staff notes that Revision 17 inadvertently revised Section 3.8.2.6 of the AP1000 DCD to specify a minimum service metal temperature of -26.1°C (-15°F). In a letter dated June 18, 2010, the applicant proposed to change the minimum service metal temperature back to -28.1°C (-18.5°F), which is supported by the bounding analysis. Therefore, the staff finds this proposed change acceptable, and identifies this as Confirmatory Item CI-SRP3.8.2-CIB1-01 to incorporate this proposed change in the AP1000 DCD.

The NRC staff reviewed the proposed changes as they relate to Revision 16 to the AP1000 DCD, Section 3.8.2.6. The proposed changes, as identified in TR-113, have been adequately incorporated into Revision 16. Accordingly, these changes are generic and are expected to be used in all COL applications referencing the AP1000 certified design.

3.8.2.6 Testing and Inservice Inspection Requirements

During the review of AP1000 DCD Tier 2, Revision 16, the staff identified that Section 3.8.2.7 had been revised to remove the requirement that the inservice inspection of the CV would be performed in accordance with the ASME Code, Section XI, Subsection IWE, and that this is the responsibility of the COL applicant. In accordance with the guidance presented in NUREG-0800 Section 3.8.2, this information should be provided by the applicant for review by the staff. Therefore, the staff requested, in RAI-SRP3.8.2-SEB1-05, that the applicant include in the AP1000 DCD information that describes how the AP1000 containment complies with the 10 CFR 50.55a requirements and the ASME Code, Section XI for the preservice and inservice examination of the containment.

In a letter dated February 27, 2009, the applicant indicated that Section 3.8.2.7 of the AP1000 DCD would be revised to reference Section 6.6, which identifies that the COL applicant will perform inservice inspection of the containment according to the ASME Code, Section XI. Section 6.6.9.1 includes a COL information item for the COL applicant to prepare preservice and inservice inspection programs for the ASME Code systems and components.

Section 6.6 was revised in the AP1000 DCD, Revision 17 to specifically include ASME Code Class MC components. The applicant indicated that Sections 6.6.9.1 and 6.6.9.2 will be revised to also specifically include Class MC systems and components.

The staff concludes that the RAI response is acceptable because: (1) the applicant will revise AP1000 DCD Section 3.8.2.7 to reference Section 6.6, which indicates that inspection of the containment is performed in accordance with the ASME Code, Section XI and 10 CFR 50.55a; (2) AP1000 DCD Section 6.6 indicates that COL applicants will prepare the inspection program for the containment; and (3) the applicant will revise AP1000 DCD Sections 6.6.9.1 and 6.6.9.2 to require the preparation of an inspection program for Class MC (containment) systems and components. Pending revision of the AP1000 DCD to incorporate the changes proposed in the RAI response, this is being tracked as Confirmatory Item CI-SRP3.8.2-SEB1-05.

3.8.2.7 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and that the applicant's application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 containment as documented in AP1000 DCD, Revision 17, against the relevant acceptance criteria as listed above and in NUREG-0800 Section 3.8.2.

When the confirmatory items described for TR-09 above are addressed, the NRC staff will be able to conclude that APP-GW-GLR-005, TR-09, "Containment Vessel Design Adjacent to Large Penetrations," is acceptable on the basis that the analyses and design were performed in accordance with the ASME Code, Section III, applicable RGs, and NUREG-0800 Section 3.8.2.

When Confirmatory Item CI-SRP3.8.2-CIB1-01, described above is addressed, the staff will be able to conclude that the revisions proposed by the applicant to AP1000 DCD, Section 3.8.2.5, meet the requirements of the10 CFR 50.55a and the ASME Code, Section III; and, therefore, are acceptable.

The staff concludes that if the items identified above are addressed, the design of the containment will continue to meet all applicable acceptance criteria.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each applicant would have to address these issues individually.

3.8.3 Concrete and Steel Containment Internal Structures

Using the regulatory guidance in NUREG-0800 Section 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," the staff reviewed: (1) description of the internal structures; (2) applicable codes, standards, and specifications; (3) loads and loading combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, and special construction techniques; (7) testing and inservice surveillance programs; (8) ITAAC; and (9) COL action items and certification requirements and restrictions. Not all of these areas were applicable to the review of the proposed changes to AP1000 Section 3.8.3 and the following SER sections provide the staff's evaluation for the relevant areas.

In its previous evaluations of AP1000 DCD, Section 3.8.3, the staff identified acceptance criteria based on the design meeting the relevant requirements in 10 CFR 50.55a, "Codes and Standards", 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records"; GDC 2, "Design Bases for Protection Against Natural Phenomena"; GDC 4, "Environmental and Dynamic Effects Design Bases"; and GDC 50, "Containment Design Basis." The staff found that the design of the AP1000 CISs was in compliance with these requirements, as referenced in NUREG-0800 Section 3.8.3 and determined that the design of the AP1000 CISs, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

In AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.3 of the certified design:

1. As a result of the extension of the AP1000 design from just hard rock sites to sites ranging from soft soils to hard rock, various seismic re-analyses of the NI structures were performed. Whereas the original design relied upon the equivalent static method of analysis for seismic loading, the re-analyses included the additional use of response spectrum and time history methods of analysis. In DCD Revision 16, Table 3.8.3-2 was revised to include the use of the response spectrum analysis for the seismic analysis of the containment internal structures. Time history analyses were used to determine maximum soil bearing pressures under the NI and, subsequent to the submittal of DCD Revision 17, to perform an updated NI stability evaluation.

2. In DCD Revision 16, the applicant removed Section 3.8.3.4.1.2 - Stiffness Assumptions for Global Seismic Analyses in the previous certified DCD. This section discussed the stiffness properties used in the seismic analyses of the containment internal structures and the auxiliary building modules. Reference was made to DCD Table 3.8.3-1, which contained the various stiffness cases for the concrete filled steel modules used for structures inside containment and the auxiliary building. This deletion of the prior text in Section 3.8.3.4.1.2 shifted the text in the sections that followed Section 3.8.3.4.1.2 (i.e., prior Section 3.8.3.4.1.3 became Section 3.8.3.4.1.2 and prior Section 3.8.3.4.1.4 became 3.8.3.4.1.3).

3. In DCD Revision 16, the applicant revised Section 3.8.3.5.7 - Design Summary Report. DCD Revision 15 indicated that the results of the evaluation would be

documented in an as-built report by the Combined License applicant. In DCD Revision 16, this was revised to state that, "The results of the evaluation will be documented in an as-built summary report." Thus the phrase "by the Combined License applicant" was removed.

4. In DCD Revision 16, the applicant revised Section 3.8.3.5.8 - Design Summary of Critical Sections, in several subsections which describe the design of different specific critical sections. This set of revisions included changes in the text portion, revisions in a number of the DCD tables, and removal of some Tier 2* information. Some of these revisions referred to Appendix H of the DCD, which is discussed below in item 5.

5. Based on the changes discussed above for DCD Section 3.8.3.5.8, the referenced DCD Appendix 3H - Auxiliary and Shield Building Critical Sections, in both DCD Revisions 16 and 17, had substantial revisions in the text, tables, and figures.

6. In DCD Revisions 16 and 17, the applicant revised Section 3.8.3.6 - Materials, Quality Control, and Special Construction Techniques. The revisions relate to the change in material for the structural modules from Nitronic 33 to Duplex 2101, and relate to the change in the industry standard from NQA-2 to NQA-1 for packaging, shipping, receiving, storage and handling of the structural modules in accordance with industry specification AISC N690.

7. In DCD Revision 17, the applicant revised Section 3.8.3.6.3 - Concrete Placement, regarding how concrete will be placed in the CA01 module inside the containment. The previous phrase in DCD Revision 15, which stated that the concrete is placed in each wall continuously from the bottom to the top was removed, and the description of the concrete placement was revised to state that concrete will be placed either through multiple delivery trunks located along the top of the wall or through windows in the module walls or pumping ports built into the module wall.

8. A new 59.5 m^3 (2100 ft^3) pressurizer is used. It has a smaller length from the outside surface of the lower head to the outside surface of the upper head. This change was made to reduce the seismic response of the pressurizer compartment.

3.8.3.1 Applicable Codes, Standards, and Specifications

During the review of AP1000 DCD Tier 2, Revision 16, the staff noted that Sections 3.8.3.2 and 3.8.4.2 describe the codes, standards, and specifications used for structural components of the AP1000. In view of the extension of the AP1000 design to soil sites, reanalysis for updated seismic spectra, design changes made to structures, and to ensure that the AP1000 meets the safety requirements in current regulatory positions, the staff, in RAI-SRP3.8.3-SEB1-01, requested that the applicant identify whether the AP1000 plant meets industry standard American National Standards Institute/American Institute of Steel Construction (ANSI/AISC)-N690-1994, Supplement 2 (2004) and the more recent versions of the applicable

American Welding Society (AWS) standards than are currently listed in AP1000 DCD, Revision 16. These references are cited in the current NUREG-0800, Section 3.8, which was issued subsequent to the license application for the AP1000 DCD, Revision 16.

In the applicant's letters dated April 3, 2009, and October 22, 2009, the applicant stated that the references to AISC-N690-1994 and the other applicable codes, standards and specifications in AP1000 DCD Sections 3.8.3.2 and 3.8.4.2 have not changed from AP1000 DCD, Revision 15 to Revision 17. The applicant indicated that the staff previously accepted the technical basis for concluding that the standards listed in AP1000 DCD Section 3.8, Revision 15 provide sufficient conservatism or equivalent levels of safety. Therefore, the applicant does not intend to evaluate conformance to later editions and revisions of these codes and standards.

Since the staff previously accepted the use of the ANSI/AISC-N690-1994 and AWS standards in the certified design as described in AP1000 DCD, Revision 15 and these standards were considered to be acceptable, subject to certain supplementary requirements as stated in AP1000 DCD Section 3.8, the staff finds that these standards are also acceptable for use in the current design of the AP1000. Therefore, RAI-SRP3.8.3-SEB1-01 is resolved.

3.8.3.2 Analysis Procedures

During the review of the AP1000 DCD Tier 2, Revision 16, the staff noted that the entire Section 3.8.3.4.1.2, "Stiffness Assumptions for Global Seismic Analyses," of the AP1000 DCD, Revision 15 had been deleted. Therefore, in RAI-SRP3.8.3-SEB1-03, the staff requested that the applicant provide a description of the CIS model, the stiffness assumptions used, and the basis for the selection of the stiffness for the CIS and auxiliary building modules.

In a letter dated February 24, 2009, the applicant provided a response, which explained that the description for the model development and analysis for the CIS are provided in AP1000 DCD Section 3.7 and TR-03. As a result of the staff's review of the RAI response, several questions were identified and these items were discussed with the applicant in a conference call on May 12, 2009. The applicant was requested to clarify the information presented in the first three rows of AP1000 DCD Table 3.8-2, regarding the specific models used. In addition, the staff requested that the applicant explain whether the models were local or global and where these analyses were described in the AP1000 DCD, and the basis for selecting the module concrete stiffness values used. During the conference call, the applicant indicated that it would provide a revised RAI response to address these items.

In a letter dated October 19, 2009, the applicant provided some information regarding the stiffness values used; however, the staff determined that further justification was needed regarding the proper stiffness utilization for the modules of the CIS and for the other RC structures. The RAI response indicates that the NI model of concrete structures is based on the gross concrete section stiffness reduced by a factor of 0.8 for the consideration of the effect of concrete cracking as recommended in Table 6-5 of FEMA 356. The staff finds that Table 6-5 of FEMA 356 indicates that the factor of 0.8 is only applicable to flexural rigidity for concrete walls that are uncracked when inspected. For walls that are cracked, the stiffness reduction factor for flexure is 0.5. For shear rigidity, the FEMA table indicates that the stiffness reduction factor is 0.4 for walls that are uncracked and cracked. Therefore, it is not appropriate to reference the FEMA standard as justification for the use of the 0.8 factor. In a follow-up RAI, the applicant

was requested to justify the stiffness reduction factor used in the analysis and design of RC structures and the concrete-filled steel members used for the CIS and other structures.

To demonstrate the adequacy of using the 0.8 stiffness reduction factor for the RC and concrete-filled steel members in the seismic analysis of the NI structures, the applicant performed a study. In a letter dated July 30, 2010, the applicant updated its responses to RAI-SRP3.7.1-SEB1-19 and RAI-SRP3.8.3-SEB1-03, and provided comparisons of the [ ] linear and [ ] nonlinear analysis results. The [ ] linear analysis used the [ ] stiffness reduction factor and the [ ] nonlinear analysis used a concrete cracking model, which reflected the concrete stiffness based on the degree of cracking in the finite elements. Both analyses were time-history analyses based on the envelope of the soil and rock profiles. Comparisons were made at the shield building roof elevation, shield building West wall (at grade elevation) and at four other locations in the auxiliary building. The response spectra at these six locations showed a reasonably close comparison to conclude that the [ ] stiffness reduction factor is acceptable.

However, the applicant did not provide [ ] comparisons for the same locations. Since [ ] is the AP1000 design basis code, the staff believes that the [ ] to [ ] comparisons are required to validate model similarity. In an updated response to RAI-SRP3.8.3-SEB1-03, dated September 3, 2010, the applicant provided the requested comparisons between the [ ] and [ ] linear analysis results. This comparison demonstrated similarity between the [ ] and [ ] models. The applicant also provided additional information on the [ ] RC to SC connection modeling approach. This information showed that the response of this [ ] RC/SC connection compared reasonably close to the detailed FEM representation of the RC to SC connection, which included the tier bars, reinforcement, steel plates, and concrete.

On the basis of the results of the studies discussed above, the staff concluded that the approach for addressing concrete cracking is acceptable. The applicant's study using [ ], supported by the correlation of linear results between [ ] and [ ], indicate that a reduced concrete modulus of [ ] is justified for the design-basis analysis of the concrete filled steel modules and RC sections., and therefore, is acceptable. The staff further concluded that the RC/SC connection simulation in the [ ] nonlinear analysis model provides a reasonable representation of the effect of the connection on the overall seismic response and its use is acceptable. Based on the DCD changes proposed in Therefore, RAI-SRP3.8.3-SEB1-03, it is classified as Confirmatory Item CI-SRP3.8.3-SEB1-03, pending revision of the AP1000 DCD to incorporate the mark-ups proposed in the RAI response.

3.8.3.3 Design Procedures and Acceptance Criteria

The staff requested, in RAI-SRP3.8.3-SEB1-04, that the applicant address concerns with the design details of the structural module connections to the RC basemat. Section 3.8.3.5.3 of the AP1000 DCD indicates that the steel plate modules are anchored to the RC basemat by mechanical connections welded to the steel plate or by lap splices. Typical details of these two options are shown on AP1000 DCD Figure 3.8.3-8, Sheets 1 and 2.

In a letter dated February 27, 2009, the applicant provided clarification of the details of the structural module connection to the basemat concrete. Correction of the connection detail on

the left side of Figure 3.8.3-8, Sheet 2, and a new alternate connection detail will be included in the next update to the AP1000 DCD. Regarding the connection detail on the right hand side of Figure 3.8.3-8, Sheet 2, the staff's understanding is that this type of connection detail is not addressed by ACI-349 Code and does not provide for a direct transfer of load from the concrete to the steel module plates as do the other two alternates. Therefore, the applicant was requested to explain why the connection detail on the right side of Figure 3.8.3-8 was not removed or to provide a technical basis to demonstrate its structural adequacy. The information provided in the RAI response, which simply made a reference to recommendations and test data given in a paper presented in a conference. In a conference call on May 12, 2009, the staff discussed the above items with the applicant, and the applicant agreed to provide a revised RAI response to address the staff's concerns.

In a letter dated March 12, 2010, a partial response was provided; however, the information still did not demonstrate the adequacy of the connection of the structural modules to the base concrete. Therefore, in a follow-up RAI, the staff indicated that, since the type of connection shown in the right side of AP1000 DCD Figure 3.8.3-8, Sheet 2, is not covered by ACI-349, the applicant should describe how the loads from the module could be properly transferred from the module to the embedded bars in the base concrete and explain how the design is performed. Also, the applicant was requested to explain why the design of the connection does not rely on the other existing option of transferring loads directly from the faceplates to the base concrete using vertical bars and mechanical connectors.

In response to the above requests, the applicant's letters dated July 30, 2010 and August 25, 2010, deleted the connection detail that does not have a direct load transfer path from the structural modules to the base concrete. In addition, a representative connection detail relying only on a direct load transfer path was proposed to be shown in AP1000 DCD Figure 3.8.3-8, Sheet 2, and all other connection alternatives would be deleted from the figure. Because the connection detail provided is identified as representative and the final design may differ to account for items, such as accessibility for inspection or ease of fabrication and construction, the applicant proposed to include another note; which states that any changes to the mechanical connection detail shall maintain a direct load path to transfer loads from both sides of the module surface plates to the vertical dowel bars in the base concrete through the use of intervening plates, mechanical connectors and welds. The staff found the RAI responses are acceptable because the representative design details proposed will provide a direct load path to transfer loads from both sides of the module surface plates to the vertical dowel bars in the base concrete. Based on the above discussion, this is identified as Confirmatory Item CI-SRP3.8.3-SEB1-04 pending revision of the AP1000 DCD to incorporate the mark-ups proposed in the RAI response.

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that AP1000 DCD, Revision 16, Tables 3.8.3-3 through 3.8.3-7 have been revised removing their identification as Tier 2*. The revised tables removed information that provided the required plate thicknesses and stress results that permit comparison to the plate thicknesses provided and allowable stress limits. In RAI-SRP3.8.3-SEB1-07, the staff requested that the applicant provide the information in the AP1000 DCD, Revision 16, for these tables equivalent to that provided in Revision 15. Also, AP1000 DCD, Revision 16, Table 3.8.3-7 replaced specific AISC interaction ratio values in Revision 15 with a notation that it is now less than 1.0 at all entries of the table. Therefore, the

staff requested that the applicant present the actual interaction ratios as was done in the prior version of the AP1000 DCD.

In a letter dated March 2, 2009, the applicant provided an explanation as to why the Tier 2* information was revised in Revision 16 of the AP1000 DCD. One explanation was that these changes were communicated to the NRC in APP-GW-GLR-045 (TR-57), Revision 1, dated November 21, 2007, Chapter 5.0, "DCD Mark Up" (November 2007), and these changes were also discussed in an audit meeting in Pittsburgh. The RAI response did not provide the requested stress results and the AISC interaction ratio values. The staff reviewed the RAI response and concluded that it did not justify the elimination of the Tier 2* designation of the design information for the critical sections. The AP1000 DCD must provide a complete design for the AP1000 plant and some of this information may be identified as Tier 2* information. In a conference call on May 12, 2009, the staff discussed these issues with the applicant, which agreed to provide a revised RAI response to address the staff's concern.

In a letter dated March 15, 2010, the applicant indicated that all of the information in Table 3.8.3-7 comparable to the data presented in the same table in the AP1000 DCD, Revision 15, will be provided on the proposed mark-ups to the AP1000 DCD amendment application. The changes to the other AP1000 DCD tables are provided in the response to RAI-SRP3.8.3-SEB1-05. The staff's review of the mark-ups for Table 3.8.3-7 concluded that the information provided is comparable to the table in the AP1000 DCD, Revision 15, and that the tabulated results for the steel wall of the IRWST show the interaction ratios are all less than 1.0 in accordance with the AISC and the ASME Code stress limits. The staff met with the applicant on October 14, 2010, to discuss the applicant's proposed identification of Tier 2* items in the proposed DCD. As a result, the applicant stated it is revising the DCD to include revised Tier 2* items in Revision 2 to the response to RAI-SRP3.8.3-SEB1-07, dated October 21, 2010. Based on the above discussion, this is identified as Confirmatory Item CI-SRP3.8.3-SEB1-07, pending revision of the AP1000 DCD to incorporate the mark-ups proposed in the RAI response.

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified several items, described in AP1000 DCD Section 3.8.3.5.8, related to the design summary of critical sections for the CIS to be addressed. These items affect Section 3.8.3.5.8.1, "Structural Wall Modules"; Section 3.8.3.5.8.2, "IRWST Steel Wall"; and Section 3.8.3.5.8.3, "Column Supporting Operating Floor." In RAI-SRP3.8.3-SEB1-05, the staff requested the applicant explain: (1) why certain Tier 2* information and criteria were removed from the AP1000 DCD; (2) why references for CIS are made to Appendix 3H, which is applicable to auxiliary and shield building critical sections; and (3) whether the existing results in Sections 3.8.1 through 3.8.5, and associated appendices reflect the latest set of updated analyses for the revised seismic loads and other loadings.

In a letter dated March 15, 2010, the applicant addressed most of the concerns identified in this RAI. The staff's review of the response noted that most of the Tier 2* information including descriptions, criteria, member forces, required plate thicknesses and stress results, removed from Section 3.8.3.5.8 of the AP1000 DCD, Revision 17, will be restored in AP1000 DCD Sections 3.8.3.5.8.1 to 3.8.3.5.8.3 and Tables 3.8.3-4 through 3.8.3-6. Therefore, in a follow-up RAI, the staff requested the applicant to include the required plate thicknesses, which were provided in the same table in the certified design presented in the AP1000 DCD, Revision 15, and to correct the designation of the Tier 2* information in AP1000 DCD Section 3.8.3.5.8.1.

In response to RAI-SRP3.8.3-SEB1-05, the applicant's letters dated July 2, and August 25, 2010, provided proposed mark-ups to AP1000 DCD Section 3.8.3.5.8, and the corresponding tables, where the required plate thicknesses were added. The staff reviewed the proposed mark-ups to the AP1000 DCD and concluded that they are acceptable because corrections were made to include the required plate thicknesses and to correct the improper designation of the Tier 2* information.

The staff reviewed the proposed mark-ups to the AP1000 DCD and concluded that they are acceptable because corrections were made to include the required plate thicknesses and to correct the improper designation of the Tier 2* information.

In addition, the applicant-proposed mark-ups included new criteria, which are tolerances on certain values designated as Tier 2*, intended to explain when changes in the values presented in the critical section Tier 2* tables must be reported to the NRC. The two new criteria presented are as follows:

(1) if a change increases or decreases the design parameters (e.g., reinforcement provided, concrete strength, or steel section size), then the change must be reported to the NRC; and

(2) if changes in the values of the loads, moments, and forces in the critical section tables that are designated as Tier 2* result in a required reinforcement (or plate thickness for the containment internal structures) increase greater than 10 percent of the provided reinforcement (or plate thickness for the containment internal structures) then the increase must be reported to the NRC.

Tier 2* information is part of the safety analysis report that cannot be changed by a license holder without prior approval. However, the criteria, proposed by the applicant for identifying when changes in values presented in the critical section Tier 2* tables do not have to be reported to the NRC, are not in compliance with the regulatory requirements of 10 CFR 52, Appendix D, Section VIII.6.a, because: (1) any changes made to the Tier 2* italicized or bracketed and asterisked text requires prior NRC approval; and (2) a generic criterion whereby changes in the loads or member forces that result in an increase in the required reinforcement (or plate thickness for modules) greater than 10 percent also need to be reported. The key is that the required reinforcement or plate thickness cannot change because if the Tier 2* information changes then criterion number (1) applies and the change must receive prior approval from the NRC. It should be noted that the proposed criteria for Tier 2* also apply to AP1000 DCD Section 3.8.5.4.4, Table 3.8.5-3, and AP1000 DCD Appendix 3H, for which the applicant also plans to use the new criteria. The staff met with the applicant on October 14, 2010, to provide this feedback. As a result, by letter dated October 21, 2010, the applicant stated it is withdrawing TR-57, and revising the DCD to include revised Tier 2* information in Revision 4 to the response to RAI-SRP3.8.3-SEB1-05, dated October 21, 2010. This is being tracked as Confirmatory Item CI-SRP3.8.3-SEB1-05, pending revision of the AP1000 DCD.

3.8.3.4 Materials, Quality Control, and Special Construction Techniques

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that AP1000 DCD Section 3.8.3.6 was revised regarding the use of different steel materials for CIS structural

modules from the previously certified AP1000 design. In RAI-SRP3.8.3-SEB1-06, the staff requested that the applicant discuss the revision of materials: (1) from [ ] grade steel plates and shapes for the modules to allow the use of other grade carbon steel plates and shapes; and (2) from [ ], stainless steel plates for the modules to [ ] stainless steel plates. The applicant was requested to explain why these materials were revised, how the new material properties compared to those of previous materials, and demonstrate that the new material properties are equivalent to, or better than, the properties used in the original analysis and design of the AP1000 CIS structures.

In letters, dated February 27, 2009 and July 2, 2009, the applicant identified the use of [ ] as acceptable carbon steel materials for use in the structural modules because these two materials are considered to have equivalent specifications commonly used for rolled shapes. The applicant also explained that the reason for replacing [ ], [ ] stainless steel plates [ ], [ ], for the modules is that [ ] material is not available in the required plate sizes. The staff found that [ ] have substantially different yield strengths, and that the two stainless materials also have different yield strengths. In addition, it is not clear which material was used in the various designs for qualifying the modules. Therefore, in a follow-up RAI, the applicant was requested to demonstrate that the alternative materials are equivalent to, or better than, those used in the original analysis and design of the modules.

In a letter dated August 31, 2009, the applicant provided information that demonstrated that the alternative materials for the structural modules are equivalent to, or better than, those used in the analysis and design. This was demonstrated for both the carbon steel and stainless steel materials, and, therefore, the staff concluded that the proposed use of these new materials is acceptable. The RAI response also provided some markups to reflect this change in the AP1000 DCD. Therefore, this is Confirmatory Item CI-SRP3.8.3-SEB1-06, pending revision of the AP1000 DCD.

3.8.3.5 Design Summary Report

In the AP1000 DCD, Revision 16, the applicant revised Section 3.8.3.5.7, "Design Summary Report." The AP1000 DCD, Revision 15 indicated that the results of the evaluation would be documented in an as-built report by the COL applicant. In the AP1000 DCD, Revision 16, this was revised to state that "The results of the evaluation will be documented in an as-built summary report." Thus, the phrase "by the Combined License applicant" was removed. The need to prepare the as-built summary report is being addressed by the applicant as an ITAAC. The staff's evaluation of the need to prepare the as-built report under an ITAAC is discussed in Section 3.8.6, "Combined License Information," in this SER.

3.8.3.6 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and that the applicant's application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 CISs as documented in the AP1000 DCD, Revision 17, against the relevant acceptance criteria as listed above and in NUREG-0800 Section 3.8.3.

The staff concludes that if the confirmatory items identified above are addressed, the design of the CISs will continue to meet all applicable acceptance criteria.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each applicant would have to address these issues individually.

3.8.4 Other Seismic Category I Structures

Using the regulatory guidance in NUREG-0800 Section 3.8.4, "Other Seismic Category I Structures," the staff reviewed areas related to all seismic Category I structures and other safety-related structures that may not be classified as seismic Category I, other than the containment and its internal structures. The specific areas of review provided in NUREG-0800 Section 3.8.4 are as follows: (1) description of the structures; (2) applicable codes, standards, and specifications; (3) loads and loading combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, special construction techniques, and QA; (7) testing and inservice surveillance programs; (8) ITAAC; and (9) COL action items and certification requirements and restrictions. Not all of these areas were applicable to the review of the proposed changes to AP1000 Section 3.8.4 and the following SER sections provide the staff's evaluation for the relevant areas. The AP1000 DCD amendment incorporates substantial changes to the shield building design, as well as additional analyses to confirm the adequacy of the design. As a result, this evaluation of the shield building replaces the evaluation in Section 3.8.4.1.1 of NUREG-1793 in its entirety, as well as changes to other portions of Section 3.8.4 relevant to the shield building.

In its previous evaluations of AP1000 DCD Section 3.8.4, the staff identified acceptance criteria based on the design meeting the relevant requirements in 10 CFR 50.55a, "Codes and Standards"; 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records"; GDC 2, "Design Bases for Protection Against Natural Phenomena"; and GDC 4, "Environmental and Dynamic Effects Design Bases." The staff found that the design of the AP1000 other seismic Category I structures was in compliance with these requirements, as referenced in NUREG-0800 Section 3.8.4 and determined that the design of the AP1000 other seismic Category I structures, as documented in the AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

In the AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.4 of the certified design:

1. As a result of the extension of the AP1000 design from just hard rock sites to sites ranging from soft soils to hard rock, various seismic re-analyses of the NI structures were performed. Whereas the original design relied upon the equivalent static method of analysis for seismic loading, the re-analyses included the additional use of response spectrum and time history methods of analysis. Appendix G of DCD Revision 17 indicates that the response spectrum analysis

was used for the 3D refined finite element model of the NI and for the analysis of the PCS valve room and miscellaneous - steel frame structures, flexible walls, and floors. Time history analyses were used to determine maximum soil bearing pressures under the NI and, subsequent to the submittal of DCD Revision 17, to perform an updated NI stability evaluation.

2. In DCD Revisions 16 and 17, the applicant revised the design and analysis procedures under Section 3.8.4.4.1 - Seismic Category I Structures. In particular, this section was revised significantly to reflect the change in the design of the shield building.

3. In DCD Revision 16, the applicant revised Section 3.8.4.5.3 - Design Summary Report. DCD Revision 15 indicated that the results of the evaluation will be documented in an as-built summary report by the Combined License applicant. In DCD Revision 16, this was revised to state that "The results of the evaluation will be documented in an as-built summary report." Thus the phrase "by the Combined License applicant" was removed.

4. In DCD Revision 16 and 17, the applicant revised Section 3.8.4.6.1.1 - Concrete, regarding the concrete material. For the shield building structure, the compressive strength of concrete was increased from 4,000 to 6,000 psi.

3.8.4.1 Description of Other Seismic Category I Structures

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that several revisions were made to AP1000 DCD Section 3.8.4.4.1 and Appendix 3H, some of which are Tier 2* information. In RAI-SRP3.8.4-SEB1-03, the staff requested that the applicant explain why these revisions have been made, demonstrate the design adequacy of these changes, and justify the removal of design information from the AP1000 DCD.

In a letter dated May 4, 2009, the applicant provided explanations of why changes were made in AP1000 DCD Section 3.8.4.4.1 and Appendix 3H. The applicant indicated that these are due to design changes to address the, "enhanced shield building design" features and these changes were already communicated to the NRC in APP-GW-GLR-045, Revision 1, which was later revised again to Revision 2. In a letter dated March 5, 2010, the applicant provided mark-ups to Appendix 3H of the AP1000 DCD, which restore some of the design information that was previously removed. The staff found that the restored information was not complete regarding identification of the required reinforcement for concrete sections, reduction in the number of critical sections evaluated, why certain loads do not appear in the load combinations, and apparent inconsistency in the allowable stress values. Therefore, in a follow-up RAI, the applicant was requested to address these items. In addition, there were a number of issues still outstanding with the changes related to the enhanced shield building design and the removal of Tier 2* information.

In response to the above requests, the applicant's letters dated July 26, 2010, and August 30, 2010, provided proposed mark-ups to AP1000 DCD, Appendix 3H, which: (1) add to the corresponding tables the required reinforcement for concrete sections and an appropriate number of critical sections evaluated; (2) present a revised table that incorporates the design

changes related to the enhanced shield building design; and (3) propose two new criteria, the same as presented in the evaluation for the response to RAI-SRP3.8.3-SEB1-05 in this SER, for identifying when changes in the values presented in the critical section Tier 2* tables must be reported to the NRC. In addition, the responses also explained that certain loads in some load combinations were excluded because the loads were not applicable to that load combination or that load combination did not govern the design. The differences in some of the tabulated allowable stress values are due to differences in the stress limit coefficients for tension and compression. The staff's review of the responses concluded that they are acceptable, in-part, because: (1) corrections were made to include the required reinforcement for concrete sections and an adequate number of critical sections were evaluated; (2) the critical section table was updated to reflect the design changes related to the enhanced shield building design; and (3) explanations were provided to justify why certain loads do not need to be considered.

Tier 2* information is part of the safety analysis report that cannot be changed by a license holder without prior approval. However, the criteria for identifying when changes in values presented in the critical section Tier 2* tables do not have to be reported to the NRC, are not in compliance with the regulatory requirements of 10 CFR 52, Appendix D, Section VIII.6.a, because: (1) any changes made to the Tier 2* italicized or bracketed and asterisked text requires prior NRC approval; and (2) a generic criterion whereby changes in the loads or member forces that result in an increase in the required reinforcement (or plate thickness for modules) greater than 10 percent also need to be reported. The key is that the required reinforcement or plate thickness cannot change because if the Tier 2* information changes then criterion number (1) applies and it must receive prior approval from the NRC. It should be noted that the proposed criteria for Tier 2* also apply to AP1000 DCD Section 3.8.5.4.4, Table 3.8.5-3, and AP1000 DCD Appendix 3H, for which the applicant also plans to use the new criteria. The staff met with the applicant on October 14, 2010, to provide this feedback. As a result, the applicant stated it is withdrawing TR-57 by letter dated October 21, 2010, and revising the DCD to include revised Tier 2* information in Revision 4 to the response to RAI-SRP3.8.4-SEB1-03, dated October 21, 2010. In this response, the applicant included new criteria on Tier 2* items in Subsection 3H.1 to be consistent with ASTM-6, "Standard Specification for General Requirements for Rolled Structural Steel Bars, Plates, Shapes, and Steel Piling," and ASTM-A480, "Standard Specification for General Requirements for Flat-rolled Stainless and Heat-Resisting Steel Plate, Sheet, and Strip." Therefore, this is Confirmatory Item CI-SRP3.8.4-SEB1-03 which is being tracked pending revision of the AP1000 DCD to incorporate the mark-ups proposed in the RAI response.

Nuclear Island Evaluation of Critical Sections Under Technical Report TR-57 and Report APP-1200-S3R-003

The applicant submitted versions of TR-57 on November 21, 2007, and July 1, 2008, to summarize the structural design and analysis of structures identified as "Critical Sections" in the CIS, auxiliary building, and enhanced shield building. The design of the critical sections for the CIS is summarized in AP1000 DCD Section 3.8.3. The design of the critical sections for the auxiliary and shield building is described in AP1000 DCD Appendix 3H, Section 3H.5. Two of the critical sections identified in Section 3H.5 are not included in Revision 0 of TR-57. According to TR-57, Revision 0, the information on the evaluation of these two sections will be provided in an update to TR-57 when the security-related assessment is more complete. Further, the information in TR-57, Revision 0, represents the results of detailed calculations and

analyses. According to the TR-57, Revision 0, the results will change slightly during the design finalization due to anticipated spectra changes resulting from resolution of the high frequency issues and plant security issues. TR-57, Revision 0, also states that small changes in modeling and updates to software may also have a minor effect on the results.

Subsequently, the applicant made further revisions to the shield building design and analyses, and submitted Revision 1 to the report. This report was later revised and completed in Revision 2, transmitted to the NRC in a letter dated July 1, 2008. TR-57, Revision 2, provides the design of five critical sections for the CIS and 12 critical sections for the auxiliary building. A brief description of the design of two critical sections associated with the enhanced shield building design is also presented. For comparison, the AP1000 DCD, Revision 17, as well as the certified design in the AP1000 DCD, Revision 15, also identifies the same critical sections for the CIS and auxiliary buildings.

In addition to TR-57, the applicant also submitted for the staff's review APP-1200-S3R-003, Revision 0, "Design for the AP1000 Enhanced Shield Building," dated August 31, 2009. The purpose of this document was to provide a separate report, which specifically describes the enhanced shield building design methodology, testing, constructability, and inspection. The enhanced shield building report includes the design of three regions/locations: shield building cylinder; shield building roof, exterior wall of the PCS water storage tank; and shield building roof, tension ring, and air inlets.

The NRC sent a letter, dated October 15, 2009, to the applicant on the results of its review of the applicant's August 31, 2009 design methodology report for the AP1000 shield building. The letter stated:

> By letter dated August 31, 2009, the applicant submitted its design methodology report for the AP1000 shield building. The U.S NRC has completed its review of that report. Based on that report and the body of technical information reviewed to date, the NRC has determined that the proposed design of the shield building will require modifications in some specific areas to ensure its ability to perform its safety function under design basis loading conditions and to support a finding that it will meet applicable regulations (i.e., 10 CFR 50.55a and 10 CFR Part 50, Appendix A (GDC 1 and 2)).
>
> Specifically, the design of the steel and concrete composite structural module (SC module) must demonstrate the ability to function as a unit during design basis events; the design of the connection of the SC module to the reinforced concrete wall sections of the shield building must demonstrate the ability to function during design basis events; the design of the shield building tension ring girder, which anchors the shield building roof to the wall, must be supported by either a confirmation test or a validated (or benchmarked) analysis method.

During the review of the August 31, 2009 report, the staff identified a potential error in the applicant's computer code, which had been used to proportion the cross-sectional strength of members involving concrete materials (basemat, CIS, auxiliary building, and the shield building). The staff informed the applicant about this concern and the staff's evaluation of the resolution

for this issue is described in Section 3.8.5 of this SER, regarding the basemat, where this item is identified in RAI-TR85-SEB1-29.

In a meeting held on November 18, 2009, with the applicant to discuss its new proposal on the design of its shield building, the staff indicated that the applicant does not appear to have implemented the 100-40-40 method for combination of the three direction seismic loading in accordance with RG 1.92, Revision 2, or the ASCE 4-98 method. The implementation of the 100-40-40 combination method is also discussed in Section 3.8.5 of this SER, regarding the basemat, where this item is identified in RAI-TR85-SEB1-27.

To address the various issues related to the use of the SC module in the shield building and the design of the connection of the SC module to the RC sections, the applicant performed additional analyses and testing and submitted a revised shield building report to the staff for review. Revision 3 to the shield building report was submitted by letter dated September 20, 2010.

The staff's evaluation and acceptance of the design of the critical sections in TR-57, as provided under the AP1000 DCD, Revision 15, was presented in NUREG-1793. However, because of changes in the design of the shield building, the number of critical sections has been increased. The staff's review of the additional critical sections associated with the shield building is provided Section 3.8.4.1.1 of the SER. By letter dated October 21, 2010, the applicant clarified the design basis for the proposed facility by deleting TR-57 and removing references to TR-57 from the DCD.

New Fuel Racks and Spent Fuel Racks - Technical Reports: TR-44 and TR-54

The applicant submitted APP-GW-GLR-026, TR-44, "New Fuel Rack Design & Structural Analysis," Revision 0, to summarize the structural/seismic analysis of the AP1000 new fuel storage racks. In addition, the applicant submitted APP-GW-GLR-033, TR-54, "Spent Fuel Storage Rack Structure/Seismic Analysis," Revision 0, to summarize the structural/seismic analysis of the AP1000 spent fuel storage racks. Subsequently, additional revisions were made to these TRs to incorporate changes made in response to RAIs regarding the structural analysis and design of the new and spent fuel racks for various loads and in response to related discussions held during several past design audits.

Section 3.8.4 of AP1000 DCD, Revisions 16 and 17 indicates that the new fuel and spent fuel storage racks are described in Section 9.1 of the AP1000 DCD. Therefore, a description of the technical information presented in the TRs and the staff's evaluation of the information in these reports are presented in Section 9.1 of this SER. The description; applicable codes, standards, and specifications; loads and load combinations; analysis and design approach; acceptance criteria; and construction of the fuel racks are evaluated in Section 9.1 of this SER, in accordance with the requirements of NUREG-0800 Section 3.8.4, Revision 2, Appendix D. Some of the key outstanding issues that were identified by the staff and evaluated in Section 9.1 of this SER include acceptable methods for evaluation of the horizontal impact forces at the top of the racks and evaluation of buckling at the bottom of the racks during liftoff caused by the seismic loading. In addition, reconciliation of the new seismic loads from the applicant's SSI reanalysis was needed.

Another issue is the evaluation of the spent fuel rack impact forces on the spent fuel pool walls. The concern is that with the reanalysis of the spent fuel racks to incorporate the updated seismic loading and revisions in the design of the racks the maximum impact force from a spent fuel rack onto the pool walls increased substantially. This issue is captured under RAI-SRP9.1.2-SEB1-06. In response to this RAI, the applicant's letter dated August 25, 2010, addressed the remaining questions regarding this issue. This response is also evaluated under Section 9.1.2 of this SER.

Design Summary Report

In the AP1000 DCD, Revision 16, the applicant revised Section 3.8.4.5.3, "Design Summary Report." The AP1000 DCD, Revision 15 indicated that the results of the evaluation would be documented in an as-built report by the COL applicant. In the AP1000 DCD, Revision 16, this was revised to state, "The results of the evaluation will be documented in an as-built summary report." Thus, the phrase, "by the Combined License applicant," was removed. Preparation of the as-built summary report is being addressed by the applicant as an ITAAC. The staff's evaluation of the need to prepare the as-built report under an ITAAC is discussed in Section 3.8.6, "Combined License Information," in this SER.

3.8.4.1.1 Shield Building

The applicant applied for an amendment to the certified design of the AP1000, an advanced, passive, pressurized-water reactor (PWR) design. The staff of the NRC has reviewed the revised design of AP1000 seismic Category I structures, including the shield building, as described in Revision 17 of the DCD. The staff applied the guidance provided in Section 3.8.4, "Other Seismic Category I Structures," Revision 3, issued May 2010, of NUREG-0800.

This evaluation of the shield building is based on key design-specific issues. These issues are outlined in NUREG-0800: (1) description of the structures; (2) applicable codes, standards, and specifications; (3) loads and loading combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, special construction techniques, and QA; (7) testing and inservice surveillance programs; (8) ITAAC; and (9) COL action items and certification requirements and restrictions.

The NRC staff issued NUREG-1793 in September 2004 and Supplement 1 in September 2005. Revision 15 of the AP1000 DCD was incorporated into Appendix D, "Design Certification Rule for the AP1000 Design," to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Subsequently, the applicant submitted Revisions 16 and 17 to the AP1000 DCD with additional modifications to the TRs that relate to the shield building:

- APP-1200-S3R-003, Revision 2, "Design Report for the AP1000 Enhanced Shield Building," dated May 7, 2010 (Shield Building Report)

- TR-85, APP-GW-GLR-044, Revision 1, "Nuclear Island Basemat and Foundation," dated February 2, 2009

- TR-03, APP-GW-S2R-010, Revision 4, "Extension of Nuclear Island Seismic Analysis to Soil Sites," dated March 22, 2010

With these revisions, the applicant is seeking to make the changes discussed below specific to the design of the shield building.

3.8.4.1.1.1 Safety Function and Description of the Shield Building

The shield building is a safety-related seismic Category I structure that provides structural and radiological shielding for the CV and radioactive systems located in the containment building; protects the containment from external events, including missiles, tornadoes, and seismic events; provides radiation shielding from nuclear materials in containment; supports the PCCWST; and provides for natural air circulation cooling for the CV.

The staff notes that the design of the shield building in the AP1000 is unique in that it is the first shield building design to include the support of the PCCWST at the apex of the building structure. The PCCWST holds 6.7 million pounds of emergency cooling water. This water load accounts for a considerable portion of the load on the roof of the shield building.

The shield building consists of cylindrical walls surrounding, and set at a distance from, the steel containment and a conical roof that supports the PCCWST over the containment. The cylindrical wall of the shield building supports both the roof and the PCCWST. The shield building wall is constructed with both conventional RC and new, first-of-a-kind SC wall modules, which make up about 75 percent of the structure. The SC modules consist of two steel faceplates and have concrete filled in between the faceplates. Shear studs anchor the concrete to the steel faceplates, and tie-bars connect the two outer faceplates together. The shield building roof, an RC structure, is connected to the cylindrical wall by the ring girder/tension ring. The auxiliary building roof and the external walls are connected to the SC cylindrical portion of the shield building. The floor slabs and interior structural walls of the auxiliary building are also structurally connected to the RC cylindrical portion of the shield building. The SC wall is attached to the top and sides of the RC wall with stepped and asymmetrical boundary conditions both in the vertical (meridional) and horizontal (hoop) directions (Shield Building Report, Figure 3.2-2). The SC module steel faceplates are not directly anchored to the RC walls. The SC wall and the RC wall are connected through mechanical connectors (Shiled Building Report, Figures 4.1-2, 4.1-3, 4.1-4, and 4.1-5), and the SC wall is also connected to the basemat reinforcement through mechanical splices.

The shield building structure has the following main features:

- a cylindrically shaped wall constructed of SC modules that are stacked vertically, welded together to form a cylinder, and filled with concrete
- an air-inlet region located above the cylindrical wall, designed to allow air flow for containment cooling during certain design basis accidents
- a conical RC roof structure with an integral RC water tank, called the PCCWST. The PCCWST contains approximately 6.7 million pounds of water.

- a ring girder tension ring consisting of a steel box girder filled with concrete, located at the intersection of the conical roof and the air-inlet region

- mechanical connections where the SC wall joins the RC wall

Cylindrical Wall. The executive summary of the Shield Building Report describes the cylindrical SC wall. Figure ES-3 shows the SC wall panel layout, [ ]. The thickness of the SC wall for the air-inlet region varies from [ ].

The free-standing vertical span of the west wall, the height from the top of the basemat to the bottom of the tension ring, is 50.6 m (166 ft, 3 in). The east part of the SC wall connects to the RC wall of the shield building (the part of the 3 foot thick wall protected by the auxiliary building structure) below the roof of the auxiliary building at El. 44.8 m (146 ft 10 in). The RC floors and walls of the auxiliary building are connected to the RC wall of the shield building and constrain lateral displacement of this wall. The height of the east wall above its SC/RC connection located below the roof of the auxiliary building is 36.4 m (119 ft, 5 in).

Air-Inlet Region. The air-inlet region at the top of the cylindrical wall of the AP1000 shield building has through-wall openings for air flow. These air-inlet openings consist of [ ] steel pipes at a downward inclination [ ] from the vertical. Center-to-center horizontal spacing of these tubes is [ ]. The air-inlet pipes are welded to the steel faceplates. Welded steel studs connect the steel pipes to the concrete.

Roof and PCCWST. The AP1000 shield building roof is a conical RC structure supported by a steel frame consisting of radial steel beams (main roof beams). Metal studs connect a steel plate to the bottom face of the conical RC roof slab. Two vertical, concentric RC walls on the roof, integral with the roof structure, define the boundaries of the PCCWST. At the center of the PCCWST on the roof is an air diffuser, or chimney, that is defined by the inner PCCWST wall.

Tension Ring. The main component of the tension ring is a rectangular, concrete-filled, closed section built of [ ] thick welded steel plates. At the top of the tension ring is a concrete-filled, triangular, closed section of steel plates. The bottom plate of this triangular section is the top plate of the tension ring. The exterior top plate of the triangular section is parallel to the roof slope, while the other top plate is perpendicular to the roof slope to support the roof slab and to anchor some of the roof's reinforcing bars. Attached to the tension ring are interior beam seats that support the radial roof framing girders. Steel plates stiffen the tension ring where these beams are seated.

SC/RC Connections. The SC wall of the shield building connects to the top of the RC basemat (El. 30.5 m (100 ft)) at the bottom of the west wall (for a span of 152.97 degrees). A short portion of the horizontal west wall connection, between azimuths 175.63 degrees and 190.00 degrees, is at El. 33.2 m (109 ft) with a vertical connection at azimuth 190.00 degrees at the transition between El. 30.5 m (100 ft) and El. 33.2 m (109 ft). The east part of the SC wall has a horizontal connection to the RC wall of the shield building below the roof of the auxiliary building at El. 44.8 m (146 ft 10 in), and vertical connections to the sides of this RC wall at azimuth 341.94 degrees, near Wall Q, from El. 30.5 m (100 ft) to El. 44.8 m (146 ft 10 in), and at azimuth 174.60 degrees, near wall N, from El. 33.2 m (109 ft) to El. 44.8 m (146 ft 10 in).

The staff finds that the description of the shield building structure, as provided in the Shield Building Report and as supplemented with design information in the responses to staff questions at the meeting on June 9-11, 2010, provides sufficient information to define the primary structural aspects and elements used by the applicant to design the structure to withstand the design-basis loads.

Using the guidance described in NUREG-0800 Section 3.8.4 and related RGs, the staff reviewed areas related to all seismic Category I structures and other safety-related structures that may not be classified as seismic Category I, other than the containment and its interior structures.

In its previous evaluation of Section 3.8.4 of the AP1000 DCD in NUREG-1793, the staff accepted the design of these structures because it met the following applicable requirements of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities":

- 10 CFR 50.55a, "Codes and Standards"
- Appendix A, "General Design Criteria for Nuclear Power Plants"
  - GDC 1, "Quality Standards and Records"
  - GDC 2, "Design Bases for Protection against Natural Phenomena"
  - GDC 4, "Environmental and Dynamic Effects Design Bases"

In Revisions 16 and 17 of the AP1000 DCD, the applicant proposed the following changes to Section 3.8.4 of the certified design:

- As a result of the extension of the AP1000 HR design to a design that includes a broader range of soil profiles, the applicant performed various seismic reanalyses of the NI structures. Whereas the original design relied upon the equivalent static method of analysis for seismic loading, the reanalyses included the additional use of response spectrum and time history methods of analysis. Appendix 3G to Chapter 3 of the AP1000 DCD, Revision 17, indicates that the RSA was used for the three-dimensional refined finite element model of the NI and for the analysis of the passive containment cooling water system valve room and miscellaneous steel frame structures, flexible walls, and floors. Time history analyses were used to determine maximum soil bearing pressures under the NI and, subsequent to the submittal of DCD Revision 17, to perform an updated NI stability evaluation.

- In DCD Revisions 16 and 17, the applicant revised the design and analysis procedures in Section 3.8.4.4.1 regarding seismic Category I structures. In particular, the applicant revised this section significantly to reflect the change in the design of the shield building. The shield building design has evolved as described primarily in the Shield Building Report.

- In DCD Revisions 16 and 17, the applicant revised Section 3.8.4.6.1.1, "Concrete." For the shield building structure, the compressive strength of concrete was increased from 4,000 psi design strength in the RC areas to 6,000 psi design strength in the SC structural modules. The applicant revised the test age of concrete from 28 days to

56 days and changed some details about the chemical composition in the Portland cement and the proportioning of the concrete mix.

- In TR-03, the applicant compared the corresponding acceleration profiles obtained from the SSI analyses for the various soil sites to the original HR acceleration profile used in thedesign of the AP1000. On the basis of this comparison, the applicant concluded that the AP1000 design is adequate for the range of soil sites considered.

- In response to questions from the staff relating to the above issues (discussed below), the applicant redesigned the shield building based on feedback from the staff transmitted in an NRC letter dated October 15, 2009. The Shield Building Report describes these design changes.

Based on its evaluation of the proposed shield building design provided in Revisions 16 and 17 to the AP1000 DCD, the staff issued RAI-SRP3.8.3-SEB1-01 asking the applicant to provide information about the design methodology and to specify which aspects of the shield building design are in accordance with ACI-349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," issued 2001, as modified by the additional criteria in RG 1.142, Revision 2, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," issued November 2001, and ANSI/AISC N690. In a letter dated August 31, 2009, the applicant submitted its design methodology report, APP-1200-S3R-003, Revision 0, "Design for the AP1000 Enhanced Shield Building." In a letter dated October 15, 2009, the NRC staff identified modifications that would be required to ensure that the shield building could perform its safety function under design-basis loading conditions and to support a finding that it meets the applicable regulations in 10 CFR 50.55a and GDC 1 and 2 in Appendix A to 10 CFR Part 50.

The letter identified the following key issues:

Detailing, Design, and Analysis

1. The applicant needs to demonstrate the adequacy of the design and detailing of the SC module to function as a fully composite unit as assumed in the applicant's design/analysis. In addition, the applicant needs to demonstrate that the SC module has sufficient ductility to survive severe earthquakes or tornado winds.

2. The SC module wall to RC wall connection is to be designed and detailed for both the RC and SC portion of the connection and supported by a basis for why the connections will carry the shield building design loads.

3. The design and analysis of the shield building tension ring (i.e., ring girder) and the air-inlet region should be supported by a validated design/analysis method (i.e., benchmarked to experimental data), or by confirmatory model tests.

Based on subsequent interactions, including meetings in December 2009 and January and February 2010, as well as telephone conferences between the NRC and the applicant, the

applicant submitted APP-1200-S3R-003, Revision 1, "Design Report for the AP1000 Enhanced Shield Building," dated March 22, 2010. Following the March submittal and after several telephone conferences between the NRC and the applicant, the applicant submitted APP-1200-S3R-003, Revision 2 (the Shield Building Report). The NRC reviewed the Shield Building Report and held a public meeting with the applicant on June 9-11, 2010. The meeting resulted in 21 items for applicant action, as summarized in an NRC memorandum dated July 19, 2010. The action items required the applicant to address design methods, analyses, and testing issues to help demonstrate the adequacy of the shield building design.

The applicant responded to 18 action items in its June 30, 2010, submittal and responded to the remaining Action Items 4 and 12 on July 23, 2010, and July 31, 2010. The applicant responded to Action Item 21 on September 3, 2010.

The applicant provided the following information in response to the action items:

- analysis methods, results, and justification for the structural demand and capacity of the shield building
- analysis and results, including stress/strain test data, and analysis of test specimens using material models in [ ]
- justification to support global stability in the design of the structure
- design approach and load path for the SC/RC connection, including justification for the shear friction capacity of the connection and any resulting design changes that were made based on the respective evaluations
- justification and qualification and production criteria for the use of mechanical splices in the design of the SC/RC connection
- analysis to support the design of the ring girder and the connection between the ring girder and air-inlet region of the SC wall, including a comparison of the cross-sectional forces between [ ] and [ ] codes to verify shear friction loads
- analysis to support the adequacy of the [ ] used at the transition of the SC wall at the air inlets from 91.4 to 137.2 cm (36 to 54 in) thickness
- evaluation of the effect of concrete cracking on the structural design

The applicant also submitted a supplemental report, "Final Shield Building In-Plane Shear Test Results," dated June 24, 2010, on the testing of the SC module under cyclic in-plane shear. Section 3.8.4.1.1.3.5 of this SER describes the staff's evaluation of this test.

3.8.4.1.1.2 Regulatory Basis

The AP1000 shield building protects the reactor and containment from exterior missiles generated by tornadoes and, thus, it is subject to impact loads. The AP1000 shield building is classified as a seismic Category I structure because it should remain functional during severe

earthquakes. Therefore, the shield building is subject to both seismic and impact loads and is designed and evaluated in accordance with the regulations and guidance as follows:

- 10 CFR Part 50.55a(a)(1) requires, "safety-related structures, systems, and components be designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with the importance of the safety functions to be performed."

- GDC 1 states, "Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed."

- GDC 2 states, "Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their functions."

- NUREG-0800 Section 3.8.4 refers to RG 1.142 and ACI-349.

- RG 1.142 endorses ACI-349 and sections of ACI-318, "Building Code Requirements for Structural Concrete and Commentary," issued 2008, as applicable codes for all seismic Category I concrete structures, including concrete shield buildings other than containment structures.

3.8.4.1.1.3 Evaluation

This evaluation is limited to the design basis of the shield building and does not address its ability to protect against a malevolent aircraft crash, which is a beyond-design-basis event evaluated under NUREG-0800 Chapter 19, "Severe Accidents."

3.8.4.1.1.3.1 Design Methodology and Process for Shield Building Design

In response to staff questions regarding the design methodology and the process for the design of the shield building, the applicant summarized its design process in a matrix in Table 1.2-1 and described it in Chapter 2 of the Shield Building Report. According to this description, the concrete design of the following areas of the AP1000 shield building falls directly within the scope of ACI-349:

- shield building roof
- knuckle region of the roof near the PCCWST wall
- compression ring
- PCCWST

The applicant designed these areas in accordance with the provisions in the established design codes by using linear elastic analysis methods. Specifically, the design for the sections in these areas is based on compliance with the ACI-349 Code, as supplemented with guidance in RG 1.142 for concrete structures. The design of the sections in these areas, which uses established design codes and analysis methods listed in Section 3.8.4 of NUREG-0800, satisfies the regulatory basis listed above and is, therefore, acceptable to the staff.

The following other areas of the shield building structure are considered as special features of the design because the applicant used SC modular wall design:

- SC cylindrical wall
- SC/RC connection
- air-inlet region

Codes and standards for the design of SC modular wall and associated structural components do not exist in the United States. Design guidelines for SC modular construction already exist in Japan, namely Japan Electric Association Code, Guideline 4618, "Technical Guidelines for Aseismic Design of Steel Plate Reinforced Concrete Structures—Buildings and Structures," issued in 2005. However, these guidelines were not specifically developed for external structures with configurations like those of the AP1000 shield building and have not been approved by the NRC. In the Shield Building Report, the applicant designated the areas of the building that use SC modular construction, which include the SC/RC connections, as special structures under ACI-349, Section 1.4.

The applicant applied the provisions of the established ACI-349 Code to the design of these special structures using linear analysis, nonlinear analysis, and testing. Specifically, the applicant applied the provisions in ACI-349 for the design of RC seismic Category I structures to the design of SC wall modules in the AP1000 shield building design. To validate the use of the code, the applicant performed nonlinear analysis and conducted a testing program to verify the behavior and determine the stiffness, strength, and ductility of proposed SC wall modules under monotonic and cyclic loads. In addition, the applicant reviewed international test data on SC wall modules (Appendix A to the Shield Building Report) to confirm the adequacy of the assumptions used by the integrated design process, such as the assumption that the SC wall modules would function as a composite unit under design-basis loads.

The integrated design process for the SC wall module uses standard methods of analysis to calculate stress demands on the shield building that meet the acceptance criteria in NUREG-0800, namely, linear elastic structural analysis. In addition, the design process uses benchmarked nonlinear analysis to confirm that cracking would not cause significant changes in the design demands; that is, changes that would lead to stresses that would invalidate the design obtained with the extension of the established code provisions.

The applicant's integrated design process also makes use of the design process for structural steel components in certain areas of the shield building. Specifically, it uses ANSI/AISC N690 in designing structural steel components of seismic Category I structures. The applicant used ANSI/AISC N690 in designing the following areas of the shield building:

- the steel roof that supports the concrete roof slab
- tension ring
- SC/RC connection

The design process uses provisions from two different design codes: ACI-349 Code for RC components, which uses an ultimate strength design approach, and ANSI/AISC N690 Standard for steel and composite components, which uses an allowable stress design approach. The use

of two different codes necessitates that the components or parts of components assessed against each code are clearly distinct and that appropriate load combinations are used for each case. The staff's review of the Shield Building Report concludes that these conditions have been met in an acceptable manner.

Based on the discussion above, the staff accepts the applicant's use of the design methodology provided in ANSI/AISC N690 Standard for structural steel components to design the shield building tension ring and the roof supporting steel beams. In addition, the staff accepts the applicant's approach of using ACI-349 as the basis for the design of the other areas, namely the shield building roof, the knuckle region of the roof near the PCCWST wall, the compression ring in the roof, and the PCCWST and walls.

The staff finds that although ACI-349 is not explicitly applicable to the SC modules, the applicant's design method, which is fundamentally based on ACI-349 and supported by confirmatory analysis and testing to confirm the adequacy of the design, is acceptable.

The staff's evaluation of the technical basis, including testing, confirmatory analysis, and design detailing, that supports this integrated design method appears in subsequent sections of this SER.

3.8.4.1.1.3.2 Design of the Shield Building

In the Shield Building Report, the applicant made significant design changes from previous versions of the design by replacing lap splices with mechanical splices at the SC-to-RC connection region between faceplates, increasing the thickness of SC module faceplates from [ ], using more ductile steel, and proposing a testing program to include testing for ductility and behavior under cyclic loads. The applicant also replaced the SC tension ring with a steel box girder, redesigned the air-inlet area with fewer through-wall openings, modified the concrete roof design from an SC module design method to an ACI-349 design method, moved SC/RC connections in the east side of the wall downward and away from the original area where the auxiliary building roof connected to the wall in order to avoid congestion and stress concentrations in the area, reduced the use of self-consolidating concrete, and redesigned the SC/RC connection to provide a direct load path. The applicant also replaced the original high-strength smooth anchor rods between the SC-to-RC basemat with #14 mild steel deformed reinforcing bars, as discussed during the meeting of June 9-11, 2010. The staff considers these changes to be significant improvements in the design of the structure to enable it to function as a unit under design-basis loads.

The staff evaluation of the applicant's analysis for the changes is provided below.

<u>Levels of Analysis</u>

The applicant's approach to developing the design basis involves three levels of analysis as described in the Shield Building Report, Section 2.6, Table 2.6-1. The three levels of analysis, with increasing levels of model refinement, are as follows:

Level 1 is used for determining the load magnitudes (seismic demands) imposed on the structure. Level 2 is used for determining the member forces and deformation demands.

Level 3 is used to assess the region with high stresses, strains, and displacements in the shield building, such as the connection regions. Linear elastic models are used at Levels 1 and 2. At Level 3, nonlinear analysis is used to confirm the results at the various levels of analysis.

The applicant used the Level 1 analysis to generate the design-basis ISRS and load magnitudes on the AP1000 NI. The applicant used the [ ] NI20 and [ ] NI10 models to develop ISRS and to design and analyze seismic Category I SSCs. In these analyses, the concrete material modulus of elasticity was reduced to 80 percent of its nominal value to account for minor concrete cracking. The applicant performed confirmatory analysis of the Level 1 analysis using the [ ] finite element analysis code. To accomplish this, the [ ] NI20 model was converted to an [ ] model with the capability to account for concrete cracking. The nonlinear concrete material parameters were benchmarked to SC element tests performed at Purdue University. Chapter 8 of the Shield Building Report describes the results of this confirmatory analysis.

The applicant used the Level 2 analysis to calculate structural design demands for the AP1000 NI. These analyses used the [ ] NI05 building model, which has a characteristic element size of 1.5 m (5 ft). In Section 2.6 of the Shield Building Report, the applicant stated that the accuracy of the NI05 model was validated by comparing the dynamic response to the [ ] NI10 model, which has a characteristic element size of 3 m (10 ft). The applicant performed confirmatory analysis of the Level 2 analysis using the [ ] finite element analysis code. The [ ] model is a highly refined model that explicitly accounts for the steel and concrete materials with separate shell and solid elements. In addition, nonlinear properties are used to characterize the concrete and steel materials. In Section 2.6 of the Shield Building Report, the applicant stated that the [ ] code was benchmarked to the Purdue University testing, as described in Chapter 7 of the Shield Building Report.

The applicant performed the Level 3 analysis to determine stresses, strains, and displacements of the critical high-stress regions in the shield building design using the [ ] finite element code and nonlinear inelastic material modeling. The concrete material parameters were benchmarked against Purdue University test results. The detailed submodels used included elements such as concrete, steel plate, studs, and [ ]. A strain-based failure criterion was selected to ensure acceptable limits under design-basis loads. Results from the Level 2 [ ] analyses are "handed-off" to the Level 3 [ ] analyses by imposing displacements at the boundary of the Level 3 analysis. The applicant described this handoff procedure in Appendix C.3 of the Shield Building Report.

The staff finds the design approach involving the three levels of analysis to determine the load magnitudes (seismic demands), the member forces, and deformation demands and including confirmatory analysis, provides a logical, reasonable, and adequate technical approach to developing the shield building design and, therefore, is acceptable.

The staff accepts the various levels of analysis involving the use of increasingly refined models to better determine element behavior under the design-basis seismic loads (SSE). The models reasonably account for material properties, and the resulting strain and stress data are confirmed under the Level 3 analysis, whereby the results from the standard linear elastic analysis models compare reasonably well with the results from the nonlinear models.

The staff finds that the approach is reasonable in that it enables the applicant to gain a better understanding of the behaviors of the structural elements of the design, particularly in the critical high-stressed regions of the structure such as the SC/RC connection. This SER provides the staff's evaluations of the results of this approach under the subsequent sections.

3.8.4.1.1.3.3 Confirmatory Analysis

In Chapter 8 of the Shield Building Report, the applicant described the approach for its benchmarking analysis methods. It should be noted that the applicant's analysis methods were not benchmarked by updating or "tuning" modeling assumptions to match any particular test. Rather, the applicant provided a confirmatory analysis, whereby it used [ ] and [ ] models to predict the behavior of various elements of the SC module and compared those results to those established using the ACI-349 design methods and SC module tests. The staff reviewed the confirmatory analysis used by the applicant to validate the predicted behavior under design-basis loads, as discussed below.

As previously stated, the applicant's design process for the shield building used standard methods of analysis that meet NUREG-0800 acceptance criteria, namely, linear elastic structural analysis, to calculate stress demands on the building. In addition, the design process uses confirmatory nonlinear analysis to confirm that concrete cracking and steel stresses would not cause significant changes in the design demands.

The applicant also described the approach for its confirmatory analysis methods in the September 3, 2010 supplement to the Shield Building Report. The applicant stated that the goal of the confirmatory process was to develop three-dimensional finite element models for SC structures that can be used to further evaluate the behavior and design of the AP1000 shield building. The applicant used the commercial finite element analysis codes [ ] and [ ] to perform the confirmatory analysis. The critical shield building areas (Section 10.2.2 of the Shield Building Report) designed using ACI-349 were modeled using a detailed Level 3 [ ] analysis for confirmatory purposes. These areas include Wall Q (Section C.6), west wall (Section 10.3 and Section C.5), air inlets (Section C.4), and Wall 5. Section 10.3 of the Shield Building Report summarizes the Level 3 analysis results for these four critical areas. Below is a summary of the applicant's confirmatory analysis methods, including development of the [ ] model, verification of the model predictions with experiments, and performance of the pushout and anchorage tests, followed by the staff's evaluation.

<u>[ ] Model Development</u>

The applicant used the commercial finite element analysis code [ ] to perform confirmatory calculations. Detailed [ ] models of several SC test specimens were developed and included important features of these modules, such as shear studs, [ ], steel plate, and concrete infill.

The steel elements were modeled in [ ] with a reduced integration solid element (C3D8R). The use of this solid element results in faster analysis running times. The nonlinear steel material properties were modeled using a multiaxial plasticity theory with von Mises yield surface, associated flow rule, and isotropic hardening. Table 8.2-1 of the Shield Building Report

provides nominal and material parameters for the steel elements for use in the Level 3 analyses. The applicant used measured material properties for the test specimens, described in Chapter 8 of the Shield Building Report.

The applicant modeled the concrete infill using C3D8R elements and a concrete damage plasticity model. This model has isotropic damage rules and can be used for modeling concrete behavior under uniaxial (compression, tension, and shear), cyclic, and multiaxial loading conditions. This model uses a compression yield surface with non-associated flow in compression. In tension, the model uses damaged elasticity concepts to model smeared cracking. The postcracking behavior depends on the tension stiffening modeling used for the concrete. The applicant analyzed three tension-stiffening models: a stress-displacement model (Figure 8.6-3) and two stress-strain models (Figure 8.6-4). As a result of the confirmatory analysis, the applicant selected the stress-strain model in Figure 8.6-4 with the lowest concrete tensile strength for the Level 3 analyses.

The applicant modeled the steel [ ] elements as fully embedded into the concrete infill and verified the approach using pushout tests. Section 8.9 of the Shield Building Report describes the results of these tests. The applicant also conducted finite element mesh sensitivity studies to confirm the adequacy of element size.

In the applicant's supplement to the Shield Building Report dated August 24, 2010, the applicant stated that a limitation of the confirmatory approach is that fracture of steel SC components (e.g., plates, studs, and [ ]) is not explicitly modeled. The applicant chose to establish acceptance criteria (strain limits), based on the guidelines in Nuclear Energy Institute (NEI) 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs" and the applicant's experimental results, for use in analysis as discussed in Section 10.1 of the Shield Building Report. Once the strains in these components exceeded these limits, the analysis results were judged to be no longer valid. In Section 10.1 of the Shield Building Report, the strain limits for steel studs were set at 5 percent while those for reinforcing bars, including [ ] and steel plates, were set at 2 percent. Subsequently, the applicant revised the strain limits on the [ ] to 1.5 percent, as noted in its August 24, 2010 supplement.

Section 10.1 of the Shield Building Report states that the tensile strain limits for the steel faceplates, 2-percent maximum membrane tensile strain, and for the steel reinforcing bars, 2-percent tensile strain, were taken to be half as large as those in NEI 07-13. Tensile strain limits in NEI 07-13 are already set to be conservatively less than the fracture tensile strain limits for steel materials. For the [ ], the final tensile strain limit chosen by the applicant, 1.5-percent strain, is also less than the [ ] strains at maximum tensile stresses shown in response to Action Item 5. The NRC staff has proposed accepting, through DG-1176, "Guidance for the Assessment of Beyond-Design Basis Aircraft Impacts," issued July 2009, the ductile material strain limits in Table 3-2 of NEI 07-13 for use in aircraft impact analyses. The staff's review of the applicant's material strain limits for steel faceplates (2 percent and [ ] (1.5 percent) finds that these limits are more conservative than those in NEI 07-13 (5 percent for SA 516 plate and 5 percent for Grade 60 reinforcing steel). Based on the conservative use of the failure criteria recommended in NEI 07-13, the staff finds the strain limits chosen by the applicant for the steel faceplates and reinforcing bars to be acceptable for use in confirmatory analysis.

For the shear connectors (studs), the applicant set the strain limit at 5 percent for the ASTM A108 Nelson studs. The staff reviewed the Nelson stud material specifications for similar studs and finds that the specifications require a minimum percentage of elongation (5.1 cm (2 in) gage length) of 20 percent for mild steel and concrete anchors. Therefore, the applicant's use of a strain limit of 5 percent is conservative, based on a comparison to 20-percent elongation over a 5.1 cm (2 in) guage length. On the basis of conservative use of a failure strain of 5 percent, the staff finds that a strain limit of 5 percent for A108 Nelson studs is acceptable for use in confirmatory analysis.

Verification with Experiments

In its letter dated August 24, 2010, the applicant stated that the modeling approach would be verified by qualitative and quantitative comparisons with experimental observation and results from large-scale tests conducted by the model developers themselves. The applicant compared the predicted shapes, rotations, and cracking pattern with those observed experimentally. The predictions were also evaluated for behavior by comparing the predicted cracking patterns, steel strains, and particularly the mode of failure with those observed experimentally. The applicant also made quantitative evaluations by comparing the predicted load-deformation responses with those measured experimentally.

As an example, the applicant showed the predicted behavior and failure mode for an out-of-plane shear specimen (a/d=3.5)[1] in Figure 2. The applicant stated that the model predicted the location and orientation of concrete cracks, the formation of concrete compressive struts between cracks, and the tensile stresses and yielding of [ ] at the crack locations.

In Figures 3 and 4, the applicant also compared predicted and measured load with midspan displacement response for two out-of-plane shear critical tests (a/d=3.5 and 2.5). The applicant stated that the model predicted the initial and postcracking stiffness with reasonable accuracy and that overall strength and failure were conservatively predicted. The applicant indicated that the models predict tie-bar plastic strains of 1.5 percent, the strain limit for these bars, at a displacement that approximately corresponds to the displacement in the test when the test specimens failed in a brittle manner. Using the above strain limits, the applicant stated that the finite element models were able to predict the behavior of SC modules in the elastic and postcracked regions of response (typically corresponding to load levels up to and beyond the SSE) with reasonable accuracy.

In reviewing the applicant's confirmatory analysis, the staff identified several concerns that were discussed at a June 9-11, 2010, meeting and resulted in action items for the applicant related to the analysis benchmarking and methodology:

- In Action Item 12, the staff asked the applicant to provide a typical load case at the SSE level and compare cross-sectional forces for both the standard [ ] Level 1 analysis and for a linear analysis with the [ ] Level 2 model.

- In Action Item 15, the staff asked the applicant to indicate the locations in the calculated load deflection curves where the 2-percent limiting strains (total strains) would occur.

[1] a/d refers to the length of spans to their depth, also referred to as shear span ratio.

- In Action Item 16, the staff asked the applicant to provide the benchmarking analysis for the [ ] models.

- In Action Item 17, the staff asked the applicant to describe the handoff procedure from the Level 2 model [ ] to the Level 3 model [ ].

The applicant responded to the above action items in its letter dated August 3, 2010. In response to Action Item 12, the applicant compared forces and moments resulting from linear analysis with the [ ] and [ ] models. Both of the models used linear material properties. Table 12-1 of the response compares the forces and moments generated by the two models based on seismic loading at the same location. Based on its review of the results in Table 12-1, the staff finds that the percentage difference in analysis results between [ ] and [ ] is less than 6 percent for axial tension (F_y) and bending moment (M_z). Therefore, based on the applicant's comparison of the results from linear analysis with [ ] and [ ], which indicates a difference of less than 6 percent for the significant cross-sectional forces, the staff finds the applicant's response to Action Item 12 to be acceptable.

In response to Action Item 15, the applicant provided load-deflection plots in Figures 4.1.1-1 and 4.1.1-2 for out-of-plane test specimens with a/d=3.5 and a/d=2.5, respectively. The plots have markings that show the location in the force-displacement curves where plastic strains of 1.5 percent and 2.0 percent occurred in the analysis with the benchmarked models.

In Figure 3-3 of its September 3, 2010, submittal, the applicant compared the maximum out-of-plane shear demand at the design-basis seismic load (SSE level) with test results (a/d=2.5) and analysis prediction. The staff reviewed the force-deflection plots and finds that comparisons of analysis and testing for the out-of-plane specimens (a/d=3.5 and 2.5) agree reasonably well with respect to stiffness for demands up to the SSE level. Based on this observation, the staff finds the applicant's response to Action Item 15 acceptable.

In response to Action Item 16, the applicant provided additional information on the benchmarking of the [ ] model. For in-plane shear on SC modules, the applicant developed a model with the same characteristics as those used in the shield building wall. The inner and outer steel plates were modeled with 3-foot elements and had a thickness of 1.9 cm (0.75 in). The applicant used the [ ] Winfrith material model and modeled the steel plate with a piecewise linear plasticity model. The model was loaded in pure shear, and the applicant verified the results against scaled Japanese test data (page 111)[1]. The applicant found that the model prediction of the in-plane shear capacity was in good agreement with the expected value.

For out-of-plane shear, the applicant performed additional [ ] confirmatory analysis. The applicant used models that had the same number of elements through the thickness of the wall as that used in the [ ] Level 2 analyses. Results of these comparisons, shown in Table 3.1 of the response to Action Item 16, indicate that the [ ] models are reasonably accurate for SSE load levels as well as for the range of applicability of the [ ] Level 3 models.

[1] Westinghouse Electric Company, "Presentation and Actions for WEC Meeting with NRC June 9 through June 11," June 30, 2010. (Agencywide Documents Access and Management System [ADAMS] Accession No. ML101940046)

For the Level 2 and 3 local models, the applicant provided an example comparison of analysis predictions for the Wall 5 location. The results appear in Figures 4.1.2-27 through 4.1.2-29. The NRC staff's review of these figures finds that the [ ] Level 2 and [ ] Level 3 models compare well for in-plane shear, out-of-plane shear, and axial tension. Based on the applicant's submittal of the [ ] benchmarking analysis, which presented benchmarking results for in-plane, out-of-plane, and Level 2 versus Level 3 models, the staff finds the applicant's response to Action Item 16 acceptable.

In response to Action Item 17, the applicant provided the steps performed to transfer analysis results from the [ ] Level 2 analysis to the [ ] Level 3 analysis, as well as the benchmarking of that procedure. The Level 2 and 3 integrated analysis includes the following steps:

(A) Identify critical regions in the shield building at the RC/SC interface and air-inlet regions.

(B) Generate the Level 2 model of the NI and shield building for the pushover confirmatory analysis, which includes models for the critical regions.

(C) Create Level 3 models for the same regions using the same cut boundary condition as in the Level 2 model.

(D) Perform the Level 2 analysis ([ ]) and extract the displacements at the cut boundaries of the critical regions.

(E) Apply the Level 2 displacements to the corresponding boundaries of the Level 3 models via shell elements that allow the coarse mesh Level 2 displacements to be interpolated and applied to the Level 3 nodes at the cut boundaries.

(F) Analyze the Level 3 models under the applied displacement boundary conditions in step (E).

To verify the adequacy of using displacements at the cut boundaries to transfer results from the Level 2 analysis to the Level 3 analysis, the applicant organized the benchmarking of this transfer method in two parts. The first part of this confirmatory analysis consisted of the following steps:

(A) Generate separate Level 2 models of the critical regions that match those for the Level 2 pushover analysis.

(B) Create Level 3 models for the same regions using the same cut boundary condition as in the Level 2 model.

(C) Apply unit loads at the boundaries of the Level 3 models to determine the stiffness of the Level 3 models for various loadings.

(D) Apply the same unit loads to the corresponding boundaries of the Level 2 models being benchmarked.

With this confirmatory analysis, the applicant assessed the relative stiffness of the Level 2 and Level 3 models. The range over which the response curves under the applied unit loads calculated with both models approximate each other identifies the range over which the two models have similar stiffness and, therefore, the range of acceptability of the handoff procedure. The applicant provided results from the confirmatory analyses in Figures 4.1.3-27 to 4.1.3-29 for Wall 5 and in Figure 4.1.3-31 for the air-inlet region. Based on the results in these figures, the staff finds that the applicant's handoff is acceptable for loads up to the SSE load level.

For the second part of the confirmatory analysis, the applicant developed an example simple shear wall model. The shear wall was loaded with three different loading cases (tension, in-plane shear, and out-of-plane bending) to verify the handoff procedure in different loading scenarios. Comparisons for axial tension (Figure 4.1.3-10), in-plane shear (Figure 4.1.3-12), and out-of-plane bending (Figure 4.1.3-15) show that the model and submodel compare reasonably well. Based on the review of the applicant's description of steps performed to transfer analysis results from the [ ] model and the verification results, the staff finds the applicant's response to Action Item 17 is acceptable.

Pushout Tests

The applicant performed pushout tests to evaluate the interaction between the [ ] that are welded to the steel plates and embedded in concrete infill. In Section 8.9 of the Shield Building Report, the applicant described the approach to conduct the confirmatory analysis for [ ]. All specimens used a [ ] pitch for stud spacing. Specimen 1 used normal concrete with two studs at [ ] spacing on each face with tie-bars in between the studs, while Specimen 2 used normal weight concrete with [ ] at [ ] spacing. Specimen 3 used self-consolidating concrete with [ ] aggregate and [ ]. Figures 8.9-4, 8.9-9, and 8.9-14 compare the analysis results (load displacement) and testing.

In Section 8.9.4 of the Shield Building Report, the applicant described the approach for modeling the [ ], as well as an evaluation of the mesh refinement. The applicant used the embedded method with [ ] concrete and shear connector elements for its simplicity and ability to capture the primary features of the load-slip displacement behavior.

The staff reviewed the applicant's analysis and testing, which provided results for the interactions between the [ ]. The staff reviewed Figure 8.9-4 and finds the applicant's recommended element size of [ ] to be acceptable for confirmatory analysis because the initial stiffness and strength of the shear connectors have a reasonable correlation to the test results.

Anchorage Test

In the Shield Building Report, the applicant performed a confirmatory analysis of an anchorage test. Although the anchorage test design represented an earlier design concept, described in

Revision 1 to APP-1200-S3R-003, the applicant felt that the comparison was still useful for confirmatory purposes. The applicant modeled the full-scale test specimen using [ ] and the concrete damage plasticity model. The mesh size for both the [ ] and the concrete elements was 3.8 cm (1.5 in). In Figure 8.10-2 of the Shield Building Report, the applicant provided a comparison of analysis and test results that shows that the Level 3 models predict reasonably well the strains in the steel faceplates and in the dowels for strains up to about 2 percent. Analysis results in Figure 8.10-6 show the location and orientation of concrete cracks and the formation of compressive struts between cracks, which provide a reasonable explanation for the observed behavior under the monotonic load conditions for the test. The staff reviewed the applicant's comparison of test results and analysis predictions and finds that the analysis results agree reasonably for the entire range of response analyzed and for the monotonic load conditions of the test. The staff notes that although the results reflect the early anchorage design, the comparison between the analysis and the test is acceptable for confirming the strains of the faceplates and the dowels. This finding only applies to the benchmarking of the finite element model for monotonic loading. The assessment of anchorage design may be found in Section 3.8.4 of this evaluation.

Confirmatory Analysis Results

Tables 10.3-2 through 10.3-5 of the Shield Building Report provide the results of the confirmatory analysis for critical areas: the air inlets, west wall, Wall Q, and Wall 5. For SSE load levels, the stress levels in the steel plates, [ ] are below the yield level for each component in the west wall, Wall Q, and Wall 5. In the air-inlet region, there is some predicted yielding of studs with a strain of 0.52 percent. However, this strain is less than the assumed failure strain of 5 percent. The staff finds that these results indicate that while there is some degree of concrete cracking predicted by the nonlinear analysis, as expected, the stresses and strains in the shield building critical areas are below yield, with the exception of some local stud yielding in the air-inlet region.

Conclusion on Confirmatory Analysis

In summary, the staff concludes that the applicant has: (1) performed testing to obtain data on the response and behavior for key failure modes of the SC wall modules; (2) developed confirmatory analysis models; (3) shown that the models predict the observed experimental behavior and response with acceptable accuracy up to the design-basis seismic load level (SSE); and (4) used the confirmatory analysis to predict stresses and strains in critical areas of the shield building for the SSE load level. Further, the staff finds that the applicant has adequately addressed the staff's concerns raised in Action Items 12, 15, 16, and 17, as identified in applicant's June 30, 2010, submittal.

Based on the above findings and the applicant's SSE load level predictions of low stress and strain values in the SC steel plates, [ ] the staff finds the applicant's confirmatory analysis approach to be acceptable. Further, the staff finds the applicant's use of the ACI-349 Code for the design of these critical sections to be acceptable.

3.8.4.1.1.3.4 Seismic Demand and Analysis Methods

Chapter 10 of the Shield Building Report describes the applicant's analyses to determine how the seismic demand that is imposed on the AP1000 NI is implemented in the design of the shield building.

The applicant used three-dimensional finite element models generated with the [ ] and [ ] codes to perform the dynamic analyses. These models comprised shell, beam, and solid elements to represent the structural geometry of the NI. For determining the design-basis FRS and demands used for structural design of the shield building, the applicant used the [ ] NI20 model to perform SSI analyses (for soil sites) and the [ ] NI10 model to analyze the HR site condition. Both models idealized the shield building wall structure with a single shell element representing the SC wall module. The staff reviewed this assumption and found it to be unsubstantiated in both TR-03 and in Revision 1 to APP-1200-S3R-003.

The staff was concerned that a single shell element would not be adequate to analyze the complex through-thickness strain gradients expected near structural discontinuities and to account for concrete cracking. Discussed below is the staff's evaluation of the applicant's method of designing the specific components of the tension ring, air-inlet region, W36 beams, conical roof, and PCCWST.

Determination of Responses to Earthquake Loads

For the design of the shield building, the applicant used response spectrum analyses and the [ ] NI05 model to perform seismic analyses. The applicant validated the [ ] NI05 model, which is a refined version of the [ ] NI10 model, against the NI10 model by comparing the mass participation by frequency of the various response modes of the structure. The NI05 model consists of a combination of shell elements, namely [ ] SHELL 45 for most of the SC wall, solid elements, beam elements, and lumped masses to represent the principal components and structures in the NI. The chosen finite elements for the SC modular wall and the overall refinement of the finite element model are adequate for the calculation of design load demands for the shield building wall for a structure with the proportions of the shield building. The input response spectra at the underside of the basemat were determined from the envelope of the response spectra for all soil cases as well as the HR case. The staff finds that the applicant has correctly applied the input spectra since the spectra envelop the range of soil conditions defined for the AP1000 plant.

For the design of the shield building roof, the applicant used equivalent static analyses with a more refined [ ] finite element model to calculate load demands for the air-inlet region, tension ring, and various structural components of the roof. Specifically, the applicant developed a highly detailed linear finite element model of the shield building structure above El. 205'. This model took advantage of the axial symmetry of the shield building above El. 205' to model only a quarter of the building. The applicant used this detailed quarter finite element model because the shield building roof required a more detailed finite element representation to properly capture the demands on each of its structural components.

The applicant then combined seismic responses (member forces and deformations) to determine the stresses in some regions of the shield building structure. The Shield Building

Report states that the responses of the shield building structure, from the three directions of seismic input, are combined by the square root of the sum of the squares (SRSS) method. However, as clarified in the response to RAI-TR85-SEB1-27, dated September 2, 2010, the applicant used the 100-40-40 method for combining the three directions of seismic responses for the shield building roof (tension ring, air-inlet region, W36 beams, conical roof, and PCCWST), the containment, and the basemat. Member forces from the shield building analyses were generated for each element or at critical cross-sections (e.g., the ring girder).

The application of the SSRS method is acceptable to the staff since this method is in accordance with NRC RG 1.92, Revision 2, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," issued July 2006. However, the applicant indicated that use of the 100-40-40 method has reduced the steel reinforcement area by 16 percent when compared to that of the SSRS method (page 3-17 of the Shield Building Report), which the staff believed should not occur when the 100-40-40 method is properly implemented. The applicant addressed this issue for the shield building and the containment in its response to RAI-TR85-SEB1-27 and for the basemat in its response to RAI-TR85-SEB1-32. These two RAIs were addressed and considered resolved. The staff's evaluation of the applicant's response regarding the implementation of the 100-40-40 method is described in Section 3.8 of the SER.

Design for Concrete Cracking and Steel and Concrete Composite Damping

The applicant stated that its SC wall module is designed in accordance with the strength method in ACI-349. The applicant used a linear elastic analysis finite element computer code, [ ], to quantify the seismic response of member forces in elements for the shield building design. In Section 10.2.1.1 of the Shield Building Report, the applicant stated that for design-basis seismic analysis (Level 1), concrete structures are modeled with linear elastic un-cracked properties with the modulus of elasticity reduced to 80 percent of its value. This reduction is made in order to reduce stiffness and to reflect the observed behavior of concrete when stresses do not result in significant cracking, as recommended in Table 6.5 of FEMA 356, "Pre-standard and Commentary for the Seismic Rehabilitation of Buildings," issued November 2000.

In Section 3.2.1 of the Shield Building Report, the applicant stated that the SC material damping is assumed to be 5 percent. The staff noted that 5 percent is appropriate for SSE demand and typically invokes a reasonably high response level that includes appreciable concrete cracking. However, the staff was concerned that a reduction factor of 0.8 and 5-percent material damping were incompatible.

In Appendix B to the Shield Building Report, the applicant provided the data on concrete cracking for the shield building (Figures B-18 through B-21) and the auxiliary building (Figures B-48 and B-49) predicted by [ ]. The applicant stated that the predicted concrete cracking for the shield building and auxiliary building was extensive. As a result, the staff could not find the justification for the assumption of a 0.8 reduction factor (for the stiffness ratio) and 5-percent material damping, given the level of cracking indicated in the [ ] analysis. To address this concern, the staff issued RAI-SRP3.7.1-SEB1-19 and requested that the applicant revise its response to RAI-SRP3.8.3-SEB1-03 as appropriate.

In a letter dated July 30, 2010, the applicant updated its responses to RAI-SRP3.7.1-SEB1-19 and RAI-SRP3.8.3-SEB1-03 and provided comparisons of the results of [ ] linear and nonlinear analyses that were time-history analyses based on the envelope of the soil and rock profiles. Comparisons were made at the shield building roof elevation, shield building west wall (at grade elevation), and four other locations in the auxiliary building.

The applicant also provided stress/strain curves for the [ ] linear and nonlinear analyses and showed that cracking was occurring under SSE loading using 5-percent structural damping. The NRC staff reviewed these results and finds the applicant's use of 5-percent structural damping acceptable based on the predictions of seismic demands sufficient to cause concrete cracking.

The staff reviewed the comparisons of ISRS for the analyzed locations and finds only minor differences in response between the [ ] linear and nonlinear models. The small differences in response suggest that the [ ] concrete stiffness reduction factor is a reasonable assumption for SSE loading. However, the applicant did not provide [ ] comparisons for the same locations. Since [ ] is the AP1000 design-basis code, the staff believes that the comparisons of [ ] to [ ]are necessary to validate model similarity.

At the August 18–20, 2010, structural audit, the applicant presented the comparison between the [ ] and [ ] linear analysis results. This comparison sufficiently demonstrated the similarity between the [ ] and [ ] models. In its letter dated September 3, 2010, the applicant updated its response to RAI-SRP3.8.3-SEB1-03 to include the comparisons to [ ].

In conclusion, the staff finds the approach for addressing concrete cracking acceptable. Further, the applicant's studies using [ ], and the correlation of linear results between [ ] and [ ] indicate that a reduced concrete modulus of [ ] and a damping value of 5 percent are justified for the design-basis analysis of the SC wall in the shield building. Therefore, the staff considers these technical issues to be resolved; further discussion appears in SER Section 3.7.2 (RAI-SRP3.7.1-SEB1-19) and Section 3.8.3 (RAI-SRP3.8.3-SEB1-03).

In a June 9-11, 2010, meeting, the NRC staff asked the applicant to address concerns about the redistribution of shield building forces resulting from concrete cracking. This item was identified as Action Item 4. To ensure that the dynamic analysis models accounted for the effects of the redistribution of forces caused by shield building concrete cracking, the staff asked the applicant to assess the effects of cracking near the base of the west wall and right above the roof at the auxiliary building. Further, the staff asked the applicant to demonstrate that for SSE-level loading, the maximum in-plane shear stresses remain within the limits allowed by ACI-349.

In its July 30, 2010, letter in response to Action Item 4, the applicant provided the requested comparisons using the [ ] (nonlinear) and [ ] (linear) analysis codes to address the extent of concrete cracking and any needed load redistribution caused by the cracking. The applicant compared concrete shear stress at various locations along the west wall at El. 100′. The results shown in Figures 4-3 through 4-6 of the letter indicate that the in-plane concrete shear stress using [ ] and [ ] remains below 600 psi for critical design locations analyzed. The applicant stated that these results demonstrate that the in-plane shear stress is below the allowable shear stress of 0.85×800 psi (680 psi) in ACI-349, Section 11.7.5.

The applicant also provided results for in-plane shear distribution at the east wall above the auxiliary building roof. Figure 4-8 provides a comparison of the [ ] and [ ] results and indicates that shear stress is below the ACI-349 allowable limit of 680 psi.

Based on a review of the applicant's [ ] and [ ] analysis results, the staff finds that the applicant's in-plane concrete shear stresses are below ACI-349 allowable limits at El. 100' and at the east wall above the auxiliary building roof and, thus, finds the results to be acceptable and in accordance with the criteria in NUREG-0800 Section 3.8.4. Therefore, the staff finds the applicant's response to Action Item 4 to be acceptable.

Thermal Loads - Concrete Shrinkage and Thermal Cycling

In both the NRC's letter of October 15, 2009, and Action Items 19 and 20 from the meeting of June 9-11, 2010, the staff raised concerns related to the need for the applicant to consider the effects of concrete shrinkage and thermal cycling loads in the design of the shield building. The staff based its concern, in part, on issues identified in a study by Oliva and Cramer, of the Structures and Materials Test Laboratory at the University of Wisconsin, entitled "Self-Consolidating Concrete: Creep and Shrinkage Characteristics," issued January 2008. The study shows that self-consolidating concrete may exhibit a higher dimension change because of creep and shrinkage than conventional concrete does under shear friction loads. In the Shield Building Report, the applicant predicted extensive vertical cracking because of thermal cycling. As a result, the staff asked the applicant to analyze how the extent of cracking and the load will be redistributed via the design of the shield building to preclude the effects of the cracking on the integrity of the structure.

In response, the applicant reevaluated the thermal shrinkage effect of the in-filled concrete in the SC wall module. After reviewing the parameters used in the thermal shrinkage and thermal cycling analyses, the applicant used a more realistic shrinkage strain value of 200 micrometers per meter (μm/m) or 200×10^{-6} μm/m (2×10^{-10} in/in). The applicant stated that the use of the shrinkage strain value indicates that no cracks occurred and the stresses produced on concrete and steel surface plates are extremely low.

The staff believes that the original applicant thermal shrinkage analysis, with the shrinkage strain of 400×10^{-6} μm/m, is conservative because it exceeds the realistic strain value of 200×10^{-6} μm/m. Further, the applicant performed a finite element model analysis using the same three-dimensional finite element model. The finite element model analysis performed was a coupled thermal-mechanical analysis using [ ] 6.9-EF1. This analysis consisted of two approaches—thermal shrinkage and thermal cycling. For thermal shrinkage, an equivalent temperature drop was simulated to produce a uniform thermal contraction in the concrete equal to 200×10^{-6} μm/m. For thermal cycling, a cyclical temperature gradient of 110°F over a 24-hour period was applied. This resulted in a maximum circumferential stress of 2.1 megapascal (MPa) (0.3 kilopound-force per square inch (ksi)) on concrete and -25.8 MPa (-3.74 ksi) on the steel surface plates. The thermal cycling analysis resulted in a maximum circumferential stress of 0.05 ksi on concrete and -2.02 ksi on the steel surface plates.

The staff reviewed the applicant's reanalysis of thermal cracking and found that the concrete strain of 400×10^{-6} μm/m is conservative and that vertical cracking is minimal; therefore, the reanalysis is acceptable.

3.8.4.1.1.3.5 Design and Testing for Ductility

In its letter of October 15, 2009, the staff stated that the applicant must demonstrate the adequacy of the design and detailing of the SC wall module to function as a fully composite unit as assumed in the design and analysis. In addition, the staff stated that the applicant must demonstrate that the SC wall module had sufficient ductility to survive earthquakes or tornado winds.

In response to this concern, the applicant made several design changes to the shield building. In the executive summary of the Shield Building Report, the applicant stated that design changes were made to the shield building to improve strength and ductility. These changes included adding [ ] connecting the surface plates to demonstrate that the structure will act as a unit under design-basis events. Further, design changes were made to the SC/RC connection, using mechanical connectors to directly transfer the forces from the SC structure to the RC structure, such that the connection will exhibit strength and ductility during seismic events. The applicant stated that the design of the critical features, such as the SC wall module, the SC/RC connection, and the tension ring/air-inlet region, was verified using benchmarked nonlinear analysis in order to demonstrate the overall strength and ductility of the AP1000 shield building. The applicant further stated that it performed benchmarked analyses (confirmatory analysis) and testing to demonstrate that the design has adequate margin to withstand the SSE in accordance with NRC regulations.

In Section 10.2 of the Shield Building Report, the applicant described the detailed analysis performed to support the basis for estimating the shield building system ductility (or drift ratio). The applicant calculated the drift ratio to access the level of system ductility provided in the shield building. The staff notes that the applicant's definition of drift ratio is the ratio of maximum displacement corresponding to a beyond-design-basis demand (e.g., review-level earthquake and the maximum displacement corresponding to the SSE-level demand. In its June 30, 2010, letter (page 63), the applicant provided an updated comparison of results shown in Table 10.2-5 of the Shield Building Report. The results were obtained using the Level 1-3 analysis models discussed in Section 3.8.4.1.1.3.2 of this SER. The applicant calculated a maximum drift ratio of 6.4 corresponding to the Level 3 analysis displacement (19.6 cm (7.7 in)) from 2.6 SSE loading divided by the SSE-level displacement (3.0 cm (1.2 in)). However, the staff was not able to correlate predicted drift ratios with system ductility. To address this concern, the staff asked the applicant to provide further clarification of its design in relation to ductility. In its response, the applicant supplemented the June 30, 2010, submittal with a letter dated September 3, 2010, which described its philosophy and approach to design and their implications to ductility.

The applicant stated that its design philosophy in relation to ductility is analogous to the "capacity design" approach in FEMA 356-2000, in which the designer identifies a ductile failure mechanism for the overall structure, designates structural fuses that will undergo inelastic deformations and dissipate energy, designs and details the fuses to prevent brittle failure modes from controlling their behavior, and designs the remaining portions of the structure with sufficient

strength to resist the force demands delivered by the fuse regions. This approach is referred to as a "strong column-weak" beam design approach in accordance with ACI-349-01, Article 21.4.2.2, for the design of moment-resisting frames.

The applicant's approach is to identify, from the results of the analysis for the calculation of member forces and through confirmatory analysis, the locations in the SC structure that are predicted to become plastic hinges (called fuses by the applicant) when subjected to earthquake forces. In the case of the shield building, this requires earthquake forces beyond the design basis seismic loads. Design detailing for the regions in the shield building assumed to be plastic hinge regions conforms to requirements in ACI-349-01, Articles 21.3.3.1-21.3.3.3, which results in shear reinforcing spacing of depth divided by [ ] maximum. This detailing is intended to prevent brittle failure modes from pre-empting the ductility of the plastic hinge regions. In regions outside of these assumed plastic hinge locations, the applicant's design conforms to Article 21.3.3.4, which requires shear reinforcement ([ ]) spaced at no farther apart than half of the depth dimension. In addition, the design for these regions also provides sufficient strength to meet the calculated design demands. Although the ductility detailing requirements in Sections 21.3 and 21.4 of ACI-349 do not apply to the shield building structure, the applicant invoked them for the analogy of the applicant's design approach to the "capacity design" approach.

Continuing its analogy to the "capacity design" approach, the applicant stated that in regions of high out-of-plane shear demand, close to supports and connections with other structures, [ ]. At the connection to the basemat, this region extends [ ] above the connection region, [ ]. In SC to RC connection regions within the auxiliary building, [ ] spacing extends beyond the connection to about [ ] above and to the side of those regions of the shield building where other structures, such as the shield building roof, attach to the SC wall. The actual distance above this SC to RC connections is, [ ].

In regions away from supports and connections, the AP1000 uses SC modules with [ ], which provides sufficient strength to meet the calculated demands.

The following is the staff's evaluation of the safety of the shield building based on the applicant's method of demonstrating that there is ductility in the design of the shield building.

The staff finds that ACI-349-01, Article 21.4.2.2, is intended for moment frame structures and is not directly applicable to cylindrical shell structures, such as the AP1000 shield building. Cylindrical shells will distribute forces in a manner that differs from a 2D or 3D framed structure. Specifically, cylindrical shells primarily resist seismic lateral loads through membrane action by a combination of in-plane shear, to resist lateral shears, with tensile and compressive forces to resist overturning moments. Furthermore, ACI-349-01 has neither provisions nor requirements for ductility detailing for unique structures, such as the shield building. The staff also finds that the calculation of member forces for the design basis seismic loads for the shield building did not involve load reductions that invoke the formation of plastic hinges for the dissipation of energy. In addition, the applicant's own design methodology for the shield building, based on

ACI-349-01, requires that shear strength capacity must be provided everywhere including the assumed hinge locations, which is done for the shield building.

Providing sufficient strength in the plastic hinge regions to meet the calculated shear demands is not a requirement for the "capacity design" approach. For the above reasons, the staff finds that the applicant's design methodology for the design of the shield building to resist seismic loads is not, in a strict sense, a "capacity design" approach.

However, the staff agrees that the inherent premise used in ACI-349, Article 21.4.2.2, of providing ductile detailing where demands are high, can be extended to a cylindrical shell if analysis has been performed to identify locations of high demands, and conservative out-of-plane shear strength to meet the calculated demands is provided elsewhere. For the AP1000 shield building, the applicant provides ductility detailing in the regions of high demands. In the regions of low out-of-plane shear demands, the applicant provides [ ] at a spacing less than one-half of the depth of the wall and conservative demand to capacity ratios (Reference September 3, 2010 submittal, Figure 4-1, and Reference June 30, 2010, submittal, Figures F1.1.2-1 to F1.2.2-16).

Also in the September 3, 2010, submittal, the applicant stated that cylindrical shells, such as the shield building wall, primarily resist seismic lateral loads through membrane action by a combination of in-plane shear, to resist lateral shear together with tensile and compressive forces to resist overturning moments. Subsequently, the applicant concluded, based on this understanding and the results of a [ ] for seismic loads greater than the design basis loads, an overall ductile failure mechanism would develop in the shield building structure with the structural fuses located in the SC portions of the shield building as designed. According to the applicant, the structural fuses have small inelastic strains and are located either close to the base of the structure, or at support points, or where there are connections to the auxiliary building.

More specifically, also in Section 2.0 of the September 3, 2010, submittal, the applicant states that the [ ] indicates that for seismic loads greater than the design basis loads, the overturning moment and base shear at the base of the structure cause either tension yielding of the steel plates in the SC portion, or tension yielding of the steel reinforcement in the RC portion of the shield building, depending on the loading combination and direction. In this submittal, the applicant also states that for loads greater than the seismic design basis loads, yielding of the steel faceplates from in-plane shear can occur for certain loading directions. Thus, the ductile failure mechanism for the overall structure is governed by the yielding of steel plates or yielding of steel reinforcement in the RC portion of the structure. The applicant then concluded that for loads greater than the design basis loads, the shield building would develop a ductile failure mechanism with structural fuses in the SC portions located as designed.

The staff evaluated the applicant's design approach of providing ductility detailing in the regions of high stresses and of providing the strength necessary to meet the design demands in the regions of low demands and finds it to be reasonable. This approach conforms to the approach in ACI-349-01, Articles 21.3 and 21.4 for moment resisting frames, for which ductility design is required by ACI-349, as opposed to structures such as the shield building structure for which ACI-349 does not have ductility provisions or requirements. The staff also finds that the shield

building structure, a complex cylindrical shell, distributes loads in a manner that differs from 2D or 3D frames and can be more uncertain. The staff finds that the shield building design provides conservative demand to capacity ratios in the regions of the wall with [ ] that can account for those uncertainties. Specifically, the calculated demand to capacity ratios for out of plane shear are for the most part less than or equal to 0.2. In addition, the regions of the wall where these demand to capacity ratios are higher than 0.2, and as high as about 0.5 in a few locations, are small in area and localized.

The staff finds that the combination of the low demand to capacity ratios for out-of-plane shears in the regions with [ ] spacing with ductility detailing in the regions of high demands provides reasonable assurance of the building safety under the design basis seismic loads by ensuring that the building has structural capacity in reserve, through a combination of structural strength and ductility, for the seismic design basis loads.

Testing for Strength, Cyclic Loading and Ductility

Section 7.11.1 of the Shield Building Report states that tests were conducted to demonstrate the cyclic behavior and ductility of the SC-portions of the shield building. [ ]. Since there are two types of shear loads (the one perpendicular to the wall, which is called out-of-plane shear, and the other along the wall in the hoop direction, which is called in-plane shear) acting concurrently and simultaneously on any point of the shield building during earthquakes, [ ]. One type of SC module is used at or near connection regions, which require high shear ductility and strength, and the proposed design and detail for that SC module was to use [ ] between faceplates, and spaced [ ] in both vertical and horizontal (hoop) directions. The other type of the SC modules is used for the remaining portion of the shield building wall with less shear ductility and strength demand, away from the connection regions, and the proposed design and detail for that SC module was to use [ ] between faceplates, [ ] in both vertical and horizontal (hoop) directions. The applicant's acceptance criteria for the ductility tests for each type of module under each kind of shear loads are listed below:

The applicant used the following acceptance criteria for the ductility tests:

Acceptance Criteria for Ductility Tests

For out-of-plane shear, ductility was to be established and measured through a loading protocol as follows:

- [ ].

– [

].

For in-plane shear

– [

].

Out-of-Plane Shear Testing To Demonstrate Ductility

The out-of-plane shear test specimen [ ] tie-bar spacing tested monotonically at shear span a/d=3.5 indicated a brittle failure mode at the load of [ ] and had less strength than the companion specimen tested monotonically at a/d=2.5, which attained a higher load [ ]. The test results for out-of-plane shear showed that the modules with [ ] [ ] failed in a brittle manner and that the case with a [is the more critical shear case.

However, the staff notes that information provided by the applicant in its supplemental letter dated September 3, 2010, Figure 3-3 indicates that there is sufficient margin between the load corresponding to the maximum SSE-level demand (approximately 80k) and the failure load of the both out-of-plane specimens [ ]. According to the applicant's design methodology this margin will be less than that shown in this figure when only the contribution of the steel is taken into account to account for tensile forces. Even for these conditions, the staff finds that there is significant margin in the specimen to preclude a brittle failure under design-basis (or SSE) loads.

The staff also finds that the tests results show that there is conservatism in the use of the ACI-349 equation for strength, $V_n=V_s+V_c$, for the AP1000 SC structure in that the design strength is bounded by the load at which brittle failure in the SC specimens occurred.

SC Modules under Cyclic Loads

For SC modules under cyclic loads, the applicant stated that the test specimen with [ ] developed its plastic moment capacity and had excellent cyclic behavior during the [ ]. Further, the applicant stated that the specimen demonstrated some strength degradation during the [ ].

The staff reviewed these test data, and concludes that the SC module attained a higher load [ ] than the specimen [ ], and attained a displacement ductility ratio (the displacement value at failure divided by the displacement value at yield) of [ ].

The applicant stated that the specimen with [ ] developed its expected shear strength of [ ] and had excellent cyclic behavior during the [

]. Some strength degradation during the [ ] cycles was observed, but the shear strength of the specimen was still greater than the expected shear strength. [

].

The staff reviewed the test data, and finds that the applicant defined the yield displacement at the point at which the specimen achieved the strength ($V_c + V_s$), which is different from the Δy definition of Δy as stated for the above module with [ ], and is incorrect for this test. By judging the hysteretic curves, this test specimen had not been loaded to sufficiently high displacements to induce yielding of the steel faceplates. Therefore, referring to the loading cycles as [ ], as stated by the applicant, is incorrect. The applicant addressed the staff's concern by removing [ ] signs from the figure in its September 3, 2010, submittal.

However, the applicant provided in the September 3, 2010 submittal on ductility, Figure 4-2, which shows the measured cyclic shear force mid-span displacement response of the specimen [ ]. The staff finds that the cyclic test response shows [ ]. Further, the out-of-plane shear strength of the non-fuse specimen under cyclic loading can still be estimated using the ACI-349 Code equations and the specimen exhibited adequate cyclic load behavior at load levels equivalent to calculated out-of-plane shear demands.

The staff finds that testing of SC wall modules with [ ] spacing did not demonstrate that the SC wall module is ductile because it did not meet acceptance criteria for ductility as proposed by the applicant.

Nonetheless, in the staff's view, the SC module [ ], although it failed in the first cycle at [ ], showed appreciable ductility and is expected, if it were tested at [ ], to result in reasonable ductility in the design. Therefore, in the staff's view, this test demonstrates that sufficient ductility capacity exists for the SC module [ ].

<u>In-Plane Shear Cyclic Testing To Demonstrate Ductility</u>

In Section 7.12 of the Shield Building Report, the applicant described the in-plane cyclic shear tests designed to demonstrate the cyclic behavior and ductility of the SC shield building design for in-plane shear loading. [

].

The staff's review of the test plan for the in-plane shear test (Section 7.12) finds that the test model and test set-up boundary conditions [ ], as shown in Figures 7.12-1 to 7.12-5, may provide additional resistance and can lead to an over-estimation of the actual strength of the SC wall module. The applicant had to terminate the test after [ ] due to laboratory safety constraints and, therefore, could not complete the ductility test. The staff believes that

cyclic loading beyond the yield point is needed to ascertain the ductility of the SC module and to observe the deterioration of the concrete between the faceplates.

In the September 3, 2010, submittal, the applicant provided a plot of the [ ] (Figure 5-2) and an envelope plot of cyclic lateral load (Figure 5-3). The applicant stated that the test results demonstrated that the SC specimen could undergo loads with acceptable deformations up to [ ] the SSE level.

The staff's review finds that the test was inconclusive with respect to demonstrating ductility. However, the applicant, in Section 5.1 of the submittal dated September 3, 2010, described tests on SC modules conducted by Ozaki et al. (2004) to supplement the basis for demonstrating ductile in-plane behavior. These tests on SC panels were performed to determine the cyclic in-plane shear and to evaluate the effects of various plate parameters, such as plate thickness and axial force. One of the test specimens, S4-00NN, was judged by the applicant to be the most relevant to the AP1000 SC module. [ ]. The ratio of shear stud spacing to plate thickness is 30 for specimen S4-00NN and 11.33 for the AP1000 SC module. Consequently, the applicant concluded that the behavior of the AP1000 SC module will be slightly better than that of the S4-00NN specimen. Specimen S4-00NN had a measured ductility value, defined as ultimate strain to yield strain, of 2.82, as shown in Figure 5-1 of the September 3, 2010, submittal.

The staff reviewed the Ozaki paper, and found that the test was properly conducted and credible. In SER Table 3.8-1, staff performed a review of the Ozaki, et al. paper to compare a few key parameters of the AP1000 design and the S4-00NN specimen. Based on this comparison, and the good agreement of SC parameters, the staff finds the applicant's use of the test data to demonstrate ductility of the SC wall to be appropriate.

Table 3.8-1. Comparison of Test Specimen of S4-00NN and AP1000 SC Module

Parameter	Test Specimen S4-00NN	AP1000 SC Module
SC wall thickness/faceplate thickness	44.4	[ ]
Stud spacing/wall thickness	0.67	[ ]
Stud spacing/plate thickness	30	[ ]
Concrete compressive strength (psi)	6,206	[ ]
Steel plate yield stress (ksi)	50.2	[ ]

The staff finds that although there were concerns regarding the test setup at Purdue, the test results indicate that the design for the in-plane shear strength criteria used ([ ]) is adequate.

In addition, the staff finds that although the Purdue test specimen was actually a framed shear wall and the stiffness of the frames was added to that of the wall during the test, the test results (reported in the Osaki paper) help assure the staff of the behavior of the SC wall module under SSE loads.

Conclusion of Design and Testing Related to Ductility and Safety of the Design

In summary, the staff finds that the purpose of shear tests is to establish the minimum shear reinforcement ([ ]) to the SC module so that it can function as a unit to resist both out-of-plane and in-plane shear forces, provide sufficient ductility (energy absorption/dissipation capability) for seismic-induced energy, and provide sufficient stiffness for the shield building to meet the allowable building drift limit. The staff finds that the tests were an acceptable basis to establish this minimum.

The staff finds that ACI-349 (Article 21.4.2.2) is intended for moment frame structures and is not directly applicable to cylindrical shell structures, such as the AP1000 shield building. Cylindrical shells distribute forces in a manner that differs from a two- or three-dimensional framed structure. However, the staff agrees that the inherent premise used in Article 21.4.2.2 (providing ductile detailing where demands are high) can be extended to a cylindrical shell if analysis has been performed to identify the locations of high demands.

Also, the staff finds that for the AP1000 shield building, the applicant provided ductility detailing in the regions of high demands. In the regions of low out-of-plane shear demands, the applicant provided conservative demand-to-capacity ratios (Figure 4-1 of its June 30, 2010, submittal). The staff finds this approach to be acceptable.

In addition, the staff finds that the AP1000 shield building design has [ ] spacing to ensure that the SC modules will function as a unit. For the regions of the SC wall with higher out-of-plane shear loads, and where yielding of the SC wall would be expected to initiate under a combination of tensile forces and out-of-plane bending for seismic loads in excess of the design-basis loads, the applicant detailed the SC modules with [ ] spacing to provide out-of-plane shear ductility. For the regions of the SC wall with low out-of-plane shear demands [ ], and the SC wall detailing does not provide out-of-plane shear ductility. In these regions, the out-of-plane shear demands calculated by the applicant are low and the SC wall modules as detailed provide conservative strength demand to capacity ratios.

For the in-plane shear test, the staff finds that the test results indicate that the design for the ACI-349 the in-plane shear strength criteria used, [ ] is adequate. The test results were inconclusive with respect to measurable ductility. However, cyclic ductility tests performed in Japan (documented in the Ozaki paper) indicate that the wall will exhibit ductile behavior under cyclic in-plane shear. On these bases, the staff concludes that the SC wall will provide adequate strength, stiffness, and ductility under design-basis (or SSE) seismic loads.

The staff finds the design for strength, stiffness, and ductility to be acceptable.

3.8.4.1.1.3.6 Design of the Steel and Concrete Composite-to-Reinforced Concrete and Basemat Connections

Section 4.1.1 of the Shield Building Report describes the design details for the revised shield building connection. The applicant stated that the steel liner plates are connected to the RC wall reinforcing bars by [ ] of the SC/RC connection (Figures 4.1-2 through 4.1-5 of the Shield Building Report). [

]. The [ ] connection is designed to the allowable working stress limits of ANSI/AISC N690 for loads in the reinforcing bars equivalent to 125 percent of the yield strength of the specimen.

In its review of the SC/RC connection design, the staff identified several concerns discussed at the June 9-11, 2010, meeting and documented as action items. In Action Item 7, the staff asked the applicant to clarify the design and load path for the SC/RC connection. In Action Item 8, the staff asked the applicant to provide justification that voids in the SC/RC connection region would not affect the load path in compression. In Action Item 9, the staff asked the applicant to provide verification that calculated shear friction values in the SC/RC connection are below the ACI-349 allowable limit. In Action Item 11, the staff asked the applicant to identify the type of [ ] connector used for the shield building, in accordance with ACI-318, Chapter 21, and to justify the use of [ ], as appropriate.

In its August 3, 2010, letter, the applicant provided responses to the above action items. In response to Action Item 7, the applicant, in Table 2.1.1-1 (page 66), stated that it would implement a design change to the SC/RC connection. The applicant stated that [ ] will be used to connect the #14 reinforcement bars in the basemat to the [ ] connection. In addition the applicant compared connection yield capacities of the SC/RC connection components, such as the [

]. In addition, the applicant summarized the stress ratio (i.e., demand to capacity ratio) for the various loading conditions on the SC/RC connection components. In Table 2.1.1-2, the applicant provided the stress ratios for tension (0.37), compression (0.84), moment (0.08), in-plane shear (0.84), out-of-plane shear (0.05), and combined tension, bending, and in-plane shear (0.64).

Based on the applicant's description and data for the SC/RC design change, component capacities, and component stress ratios that are all less than one, the staff finds the applicant's response acceptable.

Further, for Action Item 7, the applicant described the load path and showed that with the combination of 2.5 cm (1 in) thick liner plate, 5.1 cm (2 in) support plate, 5.1 cm (2 in) gusset plate, [ ], the RC/SC connection can transfer loads from tension, compression, bending moments, and shear. Hence, the load path is established through the SC/RC connection and is acceptable to the staff.

In response to Action Item 8, the applicant stated that small gaps under the connection support plates will not affect the load transfer in compression. The applicant stated that the gap under the support plates is considered for the calculation of the capacity of the connection for compression forces, as shown in Figure 2.1.2-1. Further, the direct transfer of compression force through the concrete is only considered in the region between the support plates. The applicant calculated a compression ratio for the concrete between the support plates to be less than one (0.84). Based on the applicant's calculation of compression ratio, which neglects the

concrete contribution beneath the support plates, the staff finds the response to Action Item 8 to be acceptable.

In response to Action Item 9, the applicant stated that since the design of the SC/RC connection was changed from smooth bars to deformed reinforcement bars, the ACI-349 Code was applicable. The applicant calculated the SC/RC shear capacity in response to Action Item 7 and provided the demand-to-capacity ratios in Table 2.1.1-2. The reported demand-to-capacity ratio for in-plane shear was 0.84 and for out-of-plane shear was 0.05. This indicates that the capacity of the connection is 16 percent higher than the demand. Based on the applicant's design change from smooth to deformed reinforcement bars and the shear capacity being within ACI-349 limits, the staff finds the response to Action Item 9 to be acceptable.

In response to Action Item 11, the applicant stated that it will use the ACI-318 Type 2 mechanical splice and revised its qualification and production criteria for the Type 2 connectors in compliance with the ASME B&PV Code, Section III, Division 2, Subsection CC, "Code for Concrete Containments," Article CC-4333. In addition, the applicant will use the reinforcement mechanical splice examination criteria as defined by Article CC-5320. Based on this change, the staff finds the response to Action Item 11 to be acceptable.

Based on the applicant's responses to the above action items, the staff considers the design of the SC/RC connection to be acceptable. The staff notes that the applicant will provide a COL information item that will address the constructability of the shield building, including the SC/RC connection. Section 3.8.6 of this SER discusses and evaluates this COL information item.

Testing of the Steel and Concrete Composite-to-Reinforced Concrete Connections

In Section 7.3 of the Shield Building Report, the applicant stated that a full-scale anchorage test was performed to demonstrate the strength and ductility of the previous SC/RC connection design and its ability to develop the steel reinforcement on either side of the connection. Although the test specimen was representative of an earlier connection design, the applicant stated that the test specimen had some similarities with the revised connection. The test was also used to benchmark the [ ] analysis code for use in detailed analysis (Section 8.10 of the Shield Building Report).

In Section 7.13 of the Shield Building Report, the applicant described the results of the anchorage tests and found that the objectives and acceptance criteria were satisfied. The test demonstrated the capability of the SC/RC connection to transfer 125 percent of strength of the [ ] and the ductility of the connection region.

The staff's review of the test results confirmed that the SC/RC connection exhibited adequate strength and ductility to transfer 125 percent of the strength of the [ ]. Although the test was representative of the previous design, the staff considers the new design to have improved capacity because the [ ] bar connects [ ] to the support and liner plates. As a result, the staff does not believe that further testing is required for the SC/RC connection.

The staff finds the applicant's design of the SC/RC connection acceptable based on the applicant's revised design, demonstration of design stresses below code-allowable limits, the

use of a [ ] mechanical [ ], and the anchorage test that involved testing of a connection with some similarities to the current design of the connection.

3.8.4.1.1.3.7 *Design of the Tension Ring and Air-Inlet Region*

Chapter 5 of the Shield Building Report describes the design of the tension ring and air-inlet structure. The tension ring is located at the interface of the SC air-inlet structures and the shield building RC roof (Figure 5.1-2 of the Shield Building Report). The top of the tension ring interfaces with the RC roof slab. The tension ring supports [ ] steel roof girders that are located under the RC roof slab. The bottom of the tension ring is attached to the air-inlet structure. The bottom of the air-inlet structure is attached to the top of the cylindrical SC wall of the shield building. The applicant revised the design of the tension ring in the Shield Building Report and reduced the air-inlet areas to provide more concrete for structural strength to the air-inlet region. The steel box girder for the tension ring consists of two closed sections, both of which are filled with concrete. The top section is triangular in cross-section and has sloping top surfaces in order to interface with the RC roof slab. The bottom section is rectangular in cross-section, with steel flanges and webs.

The air-inlet structure is an SC structure [

].

The top of the faceplates of the air-inlet structure [

]. The steel faceplates are connected together by [ ] vertical spacing. The air-inlet structure is an SC structure with through-wall openings for air flow. The air-inlet pipes are connected to the infill concrete by welded shear studs on their outside surface. The air-inlet openings consist of [

]. The air-inlet pipes, spaced at approximately [

] is poured into the air-inlet structure between the faceplates and bonds to the [ ] of the faceplates and the [ ] of the air-inlet pipes. That bonding makes the air-inlet SC structure act as a unit. The [ ] thick steel plates on each face, aligned with the inner and outer flanges of the tension ring, serve as primary reinforcement. The concrete infill is connected to these steel plates with [ ]. The steel face plates at the top of the air-inlet structure [

] on the underside of the bottom tension ring web plate also function to attach the tension ring to the air-inlet structure. The faceplates at the bottom of the air inlets structure are welded to the faceplates of the SC wall.

The staff finds that the applicant's changes in the design of the tension ring girder, from an [ ], have resulted in a much improved design primarily because the design change makes the tension ring girder consistent with proven methods in ANSI/AISC N690. This change also provides a more predictable load path and stiffens the tension ring structure.

The tension ring is designed as a [ ], according to the design of the member forces in ANSI/AISC N690, and the concrete infill is credited only for stability of the steel plates.

The design loads for the tension ring and air-inlet structure are established from the [ ] linear analysis. The tension ring is designed to have high stiffness and to remain elastic under required load combinations. The air-inlet structure was designed as an SC module.

In Section 5.1 of the Shield Building Report, the applicant stated that the current plan for construction of the air-inlet structure and tension ring is for the structures to be [

] below the bottom of the tension ring.

As a result of its review, the staff raised a concern with the applicant (Action Item 13) that a construction joint in the air-inlet region [ ] below the tension ring would reduce the shear capacity of the concrete in this critical section. During construction, [ ] is poured through the holes in the horizontal web plate, and it is expected that the [ ] would flow and fill up to the top of the construction joint. The staff questioned whether the construction method for the tension ring girder/air-inlet region would disrupt the integrity of the structure and whether it would function as designed under design-basis loads.

In its June 30, 2010, letter response (page 93), the applicant provided a calculation to address shear friction loads at the air-inlet connection and construction joint in the tension ring. The applicant calculated the shear capacity of the air-inlet connection (based on ACI-349) to be [

]. As a result, the applicant concluded that the capacity of the construction joint is governed by the shear transfer at the plate at the bottom of the ring girder-to-wall interface and not by shear transfer at the plane at the construction joint. The applicant also stated that this construction joint will be prepared by intentional roughening, per the requirements of ACI-349, Article 11.7.9.

The applicant also performed a calculation for the capacity of the shear ties to show that they are adequate to address the tapered transition from the [ ] thick SC wall to a [ ] thick air-inlet wall (page 96 of the June 30, 2010, letter). The calculation assumed an axial force demand of [ ] coupled with [ ] acting in tension (lower end of the taper) and [ ] acting in compression (upper end of the taper). The applicant assumed that over a height of 0.61 m (2 ft), the [ ] have a capacity [ ]. At the elevation of the transition, the maximum out-of-plane shear [ ]. As a result, the applicant stated that the [ ] can be credited for both tension caused by the inner plate transition and the out-of-plane shear demand. At the top of the transition, the applicant calculated a maximum compressive force [ ], resulting in a [ ] demand [ ].

The staff reviewed the results of these calculations and finds that the calculations' assumptions and technical bases are based on ANSI/AISC N690 and the criteria in ACI-349, and are, therefore, acceptable for the design of the tension ring and air-inlet region of the shield building. However, the staff notes that in the June 30, 2010, letter response (page 96); the applicant stated that because of the amount of congestion in this area, constructability studies are being performed. These studies will evaluate whether the current tie-bar configuration is adequate for concrete placement and will provide insight into design details that would enhance the design. During final design detailing, the applicant will consider increasing tie-bar capacity in this region based upon the results of the constructability studies.

As discussed in "Determination of Responses to Earthquake Loads" in Section 3.8.4.1.1.3.4 of this SER, the applicant did not properly implement the 100-40-40 combination method for seismic loading from the three earthquake directions (x, y, and z) when designing the tension ring and air-inlet regions. The applicant addressed this issue in its response to RAI-TR85-SEB1-27. Section 3.8 of the SER for the AP1000 DCD describes the staff's evaluation of the applicant's response about the implementation of the 100-40-40 method. The applicant's draft response to RAI-TR85-SEB1-27, transmitted on September 23, 2010, provided tabulations for the air-inlet region and tension ring to demonstrate the adequacy of the design using the applicant's 100-40-40 method. The staff's review of these tabulations determined that the applicant's 100-40-40 method results in lower member demands than the SRSS approach (the accepted method in RG 1.92). However, there were still substantial margins when the required member demands using the SRSS combination method were compared to the provided reinforcement for the air-inlet region and to the stress allowable values for the tension ring.

Based on the staff's review of the applicant's detailed design and analysis of the tension ring and air-inlet region as discussed above, the staff finds the design of the tension ring and air-inlet region to be acceptable.

3.8.4.1.1.3.8 Design of Roof and Tank Support

The cylindrical section of the shield building structurally supports the roof, which includes the PCCWST. The PCCWST has a stainless steel liner that provides a leak-tight barrier on the inside surfaces of the tank. The shield building PCCWST and the shield building roof are designed as RC sections in accordance with ACI-349. One of the significant loads on the PCCWST roof, and supporting shield building walls, is the seismic loading. To determine the seismic loading on the PCCWST, specific procedures need to be considered. The Shield Building Report indicates that the analysis and design took into account hydrodynamic loads (caused by sloshing during a seismic event) on the PCCWST walls. Detailed calculations were performed in accordance with the procedure described in ASCE 4-98. The finite element model considered the seismic loading of the water, which consists of the impulsive mode (effective fluid weight that acts as a rigid mass) and the convective mode (effective fluid weight that represents the sloshing mass).

Since the mass of water at the top of the shield building is significant, and to ensure that the seismic hydrodynamic loading of the water was properly considered in the analysis and design of the PCCWST and the shield building structural supporting members, the staff asked the applicant to describe in greater detail its method for calculating the seismic loading. Action

Item 21 in the June 30, 2010, submittal asked the applicant to describe: (1) how it determined the seismically-induced pressure distributions of the water in the tank; (2) the maximum sloshing height of the water surface; (3) how it considered the potential sloshing impact forces on the tank roof; and (4) how it determined the maximum deflections of the supporting beams to the shield building roof and tank in order to demonstrate that these deflections meet code deflection limits.

In the RAI response, dated September 3, 2010, the applicant provided information to address the seismic-induced pressure distributions, sloshing height, and deflections of the supporting beams to the shield building roof and tank. Based on the staff's independent calculation, the staff found acceptable: (1) the magnitude of the hydrodynamic pressure at the bottom of the outer tank wall used to determine the hoop stress in the tank wall; (2) the hydrodynamic base shear used to calculate the shear stress in the tank wall; (3) the hydrodynamic moment on a section immediately above the tank base used to calculate the axial stress in the tank wall; (4) the hydrodynamic moment on a section immediately below the tank base used to design the tank supporting structure; and (5) the calculation of the water sloshing height used to ensure the water does not impact the tank roof. In addition, the maximum deflection of the supporting radial beams was within code limits. As a result, Action Item 21 is resolved and the design of the PCCWST is acceptable to the staff.

3.8.4.1.1.3.9 Use of Self-Consolidating Concrete

One of the staff's key issues, as identified in its October 15, 2009, letter, was that the applicant consider the self-consolidating concrete material properties and their effects (i.e., higher shrinkage and creep strains, less shear resistance and ductility) when compared to those of standard concrete. In its response, the applicant stated that in the Shield Building Report the use of self-consolidating concrete in the shield building would be limited to selected regions of the structure, including the knuckle regions of the roof, the tension ring, the air inlets, and selected portions of the SC-to-RC connection. Other portions of the structure would be constructed of standard concrete. Both the standard concrete and the self-consolidating concrete would have a compressive strength of f'c=6,000 psi. The applicant stated that standard concrete will be used in most parts of SC construction, with limited use in a few congested areas. The applicant addressed concrete placement, shrinkage, and creep characteristics of the concrete and their effects on the shield building design.

The predicted compressive stress in the steel plate from concrete shrinkage would be 9,000 psi, and the stress in the concrete would be 387 psi. The concrete stress is slightly higher than $4\sqrt{f'c}$ = 310 psi. However, this is a very conservative estimate because the elastic modulus is lower and there is significant tensile creep at early ages when the shrinkage rate is largest. During the meeting on June 9-11, 2010, the staff asked the applicant, in Action Item 10, to further clarify the use of [ ] and the specific locations where it will be used for the shield building. In response to the action item, in its letter of June 30, 2010, the applicant stated that [ ] is used in select locations in the enhanced shield building where access is limited for a vibrator. The applicant also specified that [ ] is to be placed in the air inlets from about El. 246' up to the top of the tension ring to about El. 83.8 ft (275 ft), and below the PCCWST from about El. 89.6 m (294 ft) to about 94.2 m (309 ft).

Based on the applicant's explanations and evaluations regarding the specific concrete strength, its properties, the considerations for limiting the placement of the [ ] only to the congested areas, and the limited use of the [ ] throughout construction of the shield building to help enhance the integrity of the structure, namely in the air inlet regions and below the PCCWST tank, the staff finds the applicant's use of [ ] to be acceptable.

3.8.4.1.1.3.10 Daily Temperature and Thermal Effects

In its October 15, 2009, letter, the staff identified an issue that the applicant had not formally addressed: the daily and seasonal thermal cycling effect on the SC modular construction. In order to address the thermal cycling effect, the applicant performed thermal analysis to quantify the effect of daily and seasonal thermal cycling on the cylindrical wall.

The applicant used a cyclical temperature gradient of magnitude 110°F over the course of 1 day to evaluate the effects of thermal cycling on the SC wall. The assumed temperature cycle is applied to the exterior shield building environment while maintaining an interior building temperature of 70°F. The result of the analysis indicated that the maximum stress in the wall is circumferential tensile stress of [ ], which is below the fatigue limit. The applicant concluded that the daily temperature cycling would not cause a fatigue problem. Based on its review of the applicant's analysis, the staff finds the applicant's evaluation of daily temperature and thermal effects acceptable.

3.8.4.1.1.3.11 Local Buckling Analysis

During its review of Revision 1 of APP-1200-S3R-003, the staff found that the applicant had not provided sufficient information to demonstrate that the SC design addressed the effects of local buckling of the SC module faceplates. In response to the staff concerns, the applicant revised the design of the SC wall module by increasing both the inner and outer plate thickness from [ ]. In Section 3.3.1 of the Shield Building Report, the applicant summarized the adequacy of surface plates to resist buckling.

The applicant assumed that the buckling modes for analysis were horizontal ripples caused by vertical loading [ ], vertical ripples caused by horizontal loading [ ], and diagonal ripples caused by in-plane shear loading [ ]. Based on these wavelengths, the applicant concluded that the longest wavelength [ ] controlled the design. The applicant assumed the plate to behave as a [ ] long column, with partial moment restraint at the ends. Appendix A to the Shield Building Report provides the empirical relationships used to evaluate the SC plate buckling capacity. The applicant referenced testing conducted to support the finding that [ ]. This buckling stress is lower than the Euler value. Using these assumptions, the applicant calculated the elastic buckling stress of [ ]. Since this buckling stress exceeds the steel plate yield stress, the applicant concluded that inelastic properties of the plate govern.

The applicant verified the performance of the steel plate under construction loads and found that the midspan deflection between [ ]. This deflection resulted in a maximum steel stress of 2.8 ksi. As a result of these small displacements and stresses, the applicant concluded that the effect of wet concrete loads on reducing buckling capacity was minimal.

In Section 3.3.1 of the Shield Building Report, the applicant stated that the compression loads in the shield building cylindrical wall are well below the strength of the section. The maximum compression is [ ].

The staff reviewed the applicant's technical basis for analyzing steel plate buckling, including empirical buckling relationships, in Appendix A to the Shield Building Report and finds the basis acceptable given the geometric similarity of the tested panels with the AP1000 design. On the basis that the applicant has performed a buckling analysis using acceptable empirical design equations and that the applicant has predicted relatively low compressive stresses from all load combinations, the staff finds the applicant's design to resist local buckling of steel plates to be acceptable.

3.8.4.1.1.3.12 Global Stability Analysis

During its review of the Shield Building Report, the staff identified that the applicant had not addressed global stability of the shield building. The global stability issue was discussed and identified under Action Item 6 at the June 9-11, 2010, meeting.

To address Action Item 6, the applicant provided an analysis of global stability in its letter dated June 30, 2010. The applicant concentrated on demonstrating that the PCCWST does not add significant weight to the structure and that the long-term effects of creep are negligible. As such, the cylindrical wall was analyzed for stability under hoop and axial compression. The applicant reported that the compressive stress resulting from the dead weight of the structure was [ ]. Consequently, the applicant stated that because the dead weight stress is small the effects of creep are negligible. The applicant performed an analysis for axial buckling and calculated that the elastic buckling compressive stress was [ ]. Because the concrete compressive stress is [ ], the applicant concluded that the concrete would crush before buckling occurred.

The staff reviewed the applicant's technical basis for global stability and found it to be consistent with the ACI Committee 334 report, "Concrete Shell Structures Practice and Commentary." The staff found the analysis to be acceptable based on an independent calculation of the critical buckling strength of elastic shells under compressive loads.

Pushover Analysis

The applicant performed nonlinear confirmatory analysis to predict the behavior of the shield building up to and beyond design basis seismic loading and assess the potential for collapse. The applicant used its [ ] model of the nuclear island to perform a nonlinear pushover analysis of the shield building. The model included the shield building and the entire

auxiliary building. This finite element model did not impose constraints that would force a mode of deformation of the shield building structure. Using this model, the applicant's analysis tracked tensile stresses and strains in the steel faceplates, in-plane and out-of-plane shear deformations and stresses, stresses and strains in the [ ], deformations in the connection regions and stresses and strains in the [ ] in the RC wall below the SC wall. The applicant's analysis explicitly modeled the interaction of the shield building with the roof and walls of the auxiliary building. The applicant's model also did not exclude the possibility of shear failures. Instead, it considered concrete cracking for out-of-plane loads as well as in-plane loads and the subsequent distribution of forces to the steel reinforcement. Since the applicant's verification and validation of the model against its own test data did not capture brittle failures, the applicant tracked the possibility of local onset of such brittle shear failures through the use of limiting strains in the [ ] as well as through the combined use of analysis methods with increasing refinement, that is, the combination of [ ] models.

For its analysis, [

]. In addition, the applicant considered various combinations of the directions and intensity of the seismic loads in the two horizontal directions and in the vertical direction. Under these loading conditions and without constraints in the response modes of the structure the applicant calculated the response of the structure to proportionally increasing loads. Proportional increase of the loads is an approximation in a static pushover analysis. As the structure yields and the response becomes increasingly inelastic, there is a potential for redistribution of the loads through the height of the structure that may affect the subsequent response mode of the structure. The results of the applicant's analysis show that significant inelastic behavior of the wall, other than concrete cracking, will not occur at the design basis loads and will only start at loads closer to the review level earthquake, On this basis, loading conditions that deviate significantly from those used by the applicant are not expected up to the SSE and RLE levels.

The applicant's analysis results showed that the highly stressed regions of the shield building were near structural discontinuities such as the connection to the basemat at the 100-ft elevation, in the region above the roof of the auxiliary building and at the connection of the SC wall to the RC walls. The analysis predicts yielding initiation through yielding of the [

].

The results of the pushover analysis confirm that the shield building stresses, strains and deformations remain small at the design basis loads and that significant yielding in the SC wall does not start until loading levels beyond the SSE and of the order of the review level earthquake. The results of the analysis confirm that the high stress areas of the wall with complex states of stress from the combination of high membrane forces and out-of-plane forces are the areas of the wall for which [ ], described in Section 4.3.5.2 of this report, showed that these models exhibit ductile out-of-plane behavior under cyclic loading.

As a result of the above global stability calculation and confirmatory pushover analysis, the staff considers the issue of global stability and related Action Item 6 to be resolved.

3.8.4.1.1.3.13 Construction and Inspection Methods

The staff had concerns about the construction and inspection methods that the applicant had planned to use to ensure the integrity and safety of the shield building design. The staff's concerns centered on the sequence of construction and considerations for the wet concrete loads, thermal loads, and welding processes to be used. The staff was also concerned about how the applicant would inspect for voids, cracking, delaminating, and substandard construction of concrete. During a meeting on February 23, 2010, the staff raised concerns related to the use of a qualified inspector in accordance with the ACI-318 Code and the need for continuous inspection throughout construction.

As indicated in Section 9.2 of the Shield Building Report, the applicant plans to construct the shield building in an alternating sequence with the construction of the CV. After setting the first ring of the CV, approximately 12.2 m (40 ft) high, the shield building modules will be installed and filled with 3 m (10 ft) concrete lifts. To help ensure the integrity of the design of the shield building, the applicant will undertake a mockup program focused on three critical areas:

(1) the vertical RC-to-SC connection
(2) the horizontal RC-to-SC connection
(3) the air-inlet/tension ring structure

The results of the mockup program will be used to gain insights into any modifications to the design that may be needed before construction.

In Section 9.5 of the Shield Building Report, the applicant specified that the welding codes and process and welding inspection criteria for structural welding are in accordance with ANSI/AISC N690 and AWS D1.1, "Structural Welding Code—Steel." In Section 9.6 of the Shield Building Report, the applicant specified that ANSI/ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," as well as ANSI/AISC N690 and AWS D1.1, govern the design requirements for the fabrication, assembly, and installation of the SC wall module components and construction inspection.

The staff is concerned that the proposed SC/RC connection and the tension ring/air-inlet connection may have constructability problems, such as steel rod alignment, aggregate size, air entrapment, and bleed water accumulation. Further, the staff is concerned that the proposed connection may have design implications, such as elongation in the reinforcing bars, shear friction transfer, and compression force transfer. The goal is to increase the confidence that the success of carefully designed mock-up tests would be replicated during construction.

In particular, the staff believes that concrete placement plans for the SC and RC connection region, tension ring and air inlet should be fully developed with emphasis on ensuring venting of air and complete filling of cavities. The applicant states in Revision 2 of the Shield Building Report that horizontal construction joints at the top of each concrete placement, including those near the bottom of the ring girder, would be prepared in accordance with ACI-349,

Article 11.7.9. Since this reference does not specify a preparation procedure, the applicant should prepare one as the construction plans progress.

With respect to staff concerns raised about the method of inspecting the SC wall module given that the design includes concrete between two steel plates without visual access, in Section 9.8 of the Shield Building Report, the applicant evaluated several nondestructive examination (NDE) technologies for their potential for determining concrete defects and proposed to use the
[

]. Although the applicant has committed to investigate other NDE technologies, the [ ] approach is acceptable to the staff when used in conjunction with acceptance criteria for defects that would trigger more detailed evaluations when necessary. In Section 9.8 of Revision 2, the applicant developed criteria for acceptable levels of defects, and in Table 9.8-2, criteria for spacing between defects.

The staff understands that the spacing of defects [ ], both the maximum spacing and the spacing used for acceptance, involve both horizontal and vertical dimensions and not just a single linear dimension. On page 33 of Revision 2, the applicant wrote that a 95/95 sampling methodology would require a random grid of 59 total sampling point locations in each of the three critical areas of the inner shield building. The staff understands this to mean that for each sampling scan in each of these critical areas, 59 sampling points would be required, and not to mean that the inspection would consist of only three scans, one per critical area, and each with 59 sampling points. The staff notes that the applicant did not provide in Revision 2 specific technologies for the more detailed evaluations when acceptance criteria are not met. Finalized inspection procedures should include those technologies. On page 9-33 of the Revision 2 report, the applicant wrote that if inspection ports cut in the steel plates become necessary for NDE, the location of those ports would be at those sample point locations. The staff notes that the Revision 2 report does not indicate if a location of inspection points is a single point location or a grid of test points. This needs to be specified in the completed inspection program.

Based on its review, the staff found that the applicant has addressed the staff's concerns. Particularly, the applicant has described the construction sequence; and the use of mock-ups in order to help ensure the integrity of the designed structure during construction. However, the staff believes that the applicant should complete its development of all construction and inspection implementation procedures, establish the QA/quality control procedures, finalize its selection of the NDE technology, and determine a method to help ensure that the results of the mock-up program and the qualification of the inspectors are implemented at the site. This topic is discussed further in Section 3.8.4.1.1.4 below.

3.8.4.1.1.4 Inspections, Tests, Analyses, and Acceptance Criteria

AP1000 DCD, Revision 17, Tier 1, Table 3.3-6 addresses the NI structures, including the critical sections. The acceptance criteria require a report that reconciles deviations during construction and concludes that the as-built shield building structures, including critical sections, conform to the design-basis loads without loss of structural integrity or the safety function. The staff finds that the AP1000 DCD Tier 1 ITAAC included sufficient requirements for the design acceptance of the shield building and its critical sections. Hence, the staff did not identify any additional ITAAC based on its review of the shield building design.

In Chapter 9 of the Shield Building Report, the applicant described the construction and inspection methods for the shield building. The staff's review found that the applicant must provide a COL information item to ensure that the shield building is constructed as designed to perform its intended safety function.

In RAI-SRP3.8.4-SEB1-04, the staff asked the applicant to provide commitments for unique construction and inspection procedures, such that the COL applicant will develop and follow procedures described in the COL information item. Further, the staff requested that the COL information item include the construction sequence, mockup requirements for the critical areas of the shield building, concrete placement methods, inspection of modules before and after concrete placement, and QA procedures.

In its response dated September 3, 2010, the applicant proposed a new COL information item including construction procedures and inspection procedures for SC construction. The applicant stated that these procedures derive from Chapter 9 of the Shield Building Report and will be added to AP1000 DCD Section 3.8. Further, the applicant stated that for SC construction, the construction inspection will be done in accordance with the applicable codes and standards listed in AP1000 DCD Section 3.8.4.2. For the shield building mockup program, the applicant proposed to use the heavily reinforced sections, which are deemed to be the sections of the design that present difficult construction issues. These sections include the lower section of RC/SC interface, horizontal RC/SC connection, and the air-inlet structure/tension ring. Additionally, the applicant stated that similar mockups will also be performed for the SC module and that insights from these mockups will be applied in construction.

The COL information item states that COL holders referencing the AP1000 DC will develop construction and inspection procedures to implement the commitments for concrete-filled steel plate modules. Further, these procedures will address concrete placement, use of construction mockups, and inspection of modules before and after concrete placement.

The staff reviewed the response to RAI-SRP3.8.4-SEB1-04 and the proposed COL information item and finds that the applicant's commitment to perform shield building mockups and develop construction and inspection procedures is acceptable. Consequently, this issue is Confirmatory Item CI-SRP3.8.4-SEB1-04 pending revision of AP1000 DCD Section 3.8.

3.8.4.1.1.5 Shield Building Conclusion

The staff evaluated the adequacy of the design of the shield building, as provided by the applicant in the Shield Building Report dated May 7, 2010, and as supplemented by submittals dated June 24, June 30, July 30, and September 3, 2010, and finds that the design of the shield building meets the relevant requirements of the regulations as provided by 10 CFR 50.55a and GDC 1 and 2 of Appendix A to 10 CFR Part 50.

Based on its evaluation, the staff finds that the design of the shield building demonstrates reasonable assurance that it will perform its intended safety function, and, therefore, is acceptable. Moreover, the staff finds that the shield building is adequately designed to withstand the effects of natural phenomena, thereby ensuring it will perform its intended safety function.

The staff recognizes that design standards or industry codes specific to the design of the SC wall module do not exist in the United States. However, the staff finds that the applicant used an alternative approach and implemented an integrated design methodology, including design, analysis, confirmatory analysis, testing, construction, and inspection, applicable for the development of the design of the AP1000 shield building. Specifically, the design methodology uses ACI-349 for RC design and supplements it with confirmatory analysis and confirmatory testing for its application to the AP1000 design of the SC wall module. Specifically, for the design of the SC cylindrical wall, air inlets, and SC/RC connection, the ACI-349 methodology was used for the design and the applicant supplemented its design with confirmatory analysis and testing. In view of the integrated methodology adopted for the shield building design, the staff believes the applicant's alternative approach is acceptable for this first-of-a-kind engineering design.

In addition, the staff finds that the applicant's modifications to improve the original design of the shield building, such as the use of the [ ] in the SC wall module and enhancements to the SC-to-RC and basemat connections, the roof, and tension ring/ring girder and air-inlet regions, make significant improvements in the design. Specifically, the applicant's inclusion of [ ] significantly improves the capacity of the SC wall module and enables the structure to function as a unit under design-basis loads. Further, the staff finds that the design possesses the basic elements of strength, stiffness, and ductility. The revised SC-to-RC connection allows for a [ ], while the revisions to the design of the tension ring and air-inlet region significantly improve the load path and thus, the transfer of forces.

The applicant's analysis of strength and ductility is acceptable for SSE demand, and the use of confirmatory tests in conjunction with confirmatory analysis demonstrates that the capacity based on ACI-349 equations for the design of SC structures is adequate to meet the SSE demands. With regard to the analysis supporting the design of the shield building, the applicant performed three levels of analysis to determine the load magnitudes, response spectra and member forces and the required design strength in accordance with the ACI-349 Code. In addition, the applicant's consideration of thermal effects, fatigue, creep, and construction loads in the design of the shield building were reasonably well supported by modeling and detailed confirmatory analyses.

As part of the integrated design methodology, the applicant conducted confirmatory tests of the SC wall module to confirm the adequacy of those portions of the AP1000 shield building design that fall outside the scope of existing design codes and to demonstrate the level of conservatism in using ACI-349. Specifically, [ ] resulted in demonstrating the desired ductile behavior, and the out-of-plane shear test with [ ]. In addition, the [ ] of the SC wall module indicated substantial strength margin to design loads, but the module was not tested to capacity; therefore, the test did not demonstrate that the SC module would not fail in a brittle manner under cyclic loading. In a report referenced by the applicant, the staff found that a Japanese test of scaled models of SC structures (with geometry similar to the AP1000 shield building design) had demonstrated sufficient ductility for cyclic in-plane shear loading. However, the Japanese tests were not performed for cyclic out-of-plane shear loading.

The applicant addressed ductility for out-of-plane loading by referencing ACI-349, Article 21, pertaining to moment-resisting frames. The staff finds that ACI-349 (Article 21) is intended for moment frame structures and is not directly applicable to cylindrical shell structures, such as the AP1000 shield building. Cylindrical shells will distribute forces in a manner that differs from a two- or three-dimensional framed structure. However, the staff agrees that the inherent premise used in ACI-349, Article 21, of providing ductile detailing where demands are high, can be extended to a cylindrical shell if analysis has been performed to identify locations of high demands.

The staff finds that to resist out-of-plane shear loading, the shield building design uses [ ] to ensure that the SC modules will function as a unit. For the regions of the SC wall module with higher out-of-plane shear loads, and where yielding of the SC wall module would be expected to initiate under a combination of tensile forces and out-of-plane bending for seismic loads, the applicant detailed the SC modules with [ ] to provide out-of-plane shear ductility. For the regions of the SC wall with low out-of-plane shear demands and [ ], the SC wall detailing does not provide out-of-plane shear ductility based on the test results. In these regions, the out-of-plane shear demands calculated by the applicant are low, and the SC wall modules as detailed provide conservative strength demand-to-capacity ratios. Based on: (1) demonstration of conservative strength and adequate cyclic behavior for the SC module with [ ] ; (2) confirmatory analysis that identified locations of potential SC steel plate yielding; and (3) the analogy with ACI-349, Articles 21.3 and 21.4, which require ductile detailing only where demands are high and plastic hinges are expected to form, the staff finds the applicant's use of [ ] at [ ] spacing to be acceptable.

Furthermore, the staff finds SC module design is acceptable on the basis that the applicant demonstrated that its lowest margin is 18 percent (in-plane shear) under design-basis SSE loads and on the staff's determination that other SC modules with design characteristics similar to the AP1000 shield building possessed sufficient ductility under in-plane shear cyclic loading. Regarding out-of-plane shear loading of the SC module with [ ], the staff finds that although these specimens failed in a brittle manner, there is significant margin between the failure loads of the two test specimens [ ] and the maximum SSE demand of [ ]. Lastly, the applicant's construction and inspection processes involving the use of mock-ups for two key areas, the SC-to-RC connection and the ring girder-to-SC connection, are acceptable, although the staff finds that the applicant should finalize its implementation of its construction and inspection procedures and methods. The applicant should also determine a method to help ensure that the results of the mock-up program are correctly implemented at the site.

In summary, based on the above discussions, the staff finds that the design of the AP1000 shield building is acceptable.

3.8.4.2 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and

that the application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 as they relate to other seismic Category I structures as documented in the AP1000 DCD, Revision 17 against the relevant acceptance criteria as listed above and in NUREG-0800 Section 3.8.4.

The staff concludes that if the items identified above are addressed, the design of the other seismic Category I structures meets all applicable acceptance criteria. In summary, based on the above discussions, the staff finds that the design of the AP1000 shield building is acceptable.

Westinghouse proposed to amend the existing design certification rule, in part, to address the requirements of the AIA rule. The AIA rule itself mandated that a DCR be revised (either during the DCR's current term or no later than its renewal) to address the requirements of the AIA rule. In addition, the AIA rule provided that any combined license issued after the effective date of the final AIA rule must reference a DCR complying with the AIA rule, or itself demonstrate compliance with the AIA rule. The AIA rule may therefore be regarded as inconsistent with the finality provisions in 10 CFR 52.63(a) and Section VI of the AP1000 DCR. However, the NRC provided an administrative exemption from these finality requirements when the final AIA rule was issued. See June 12, 2009; 74 FR 28112, at 28143-45. Therefore, the NRC has already addressed the finality provisions of applying the AIA rule to the AP1000 with respect to the AP1000 and referencing COL applicants.

3.8.5 Foundations

Using the regulatory guidance in NUREG-0800 Section 3.8.5, "Foundations," the staff reviewed areas related to the foundations of all seismic Category I structures. The specific areas of review provided in NUREG-0800 Section 3.8.5 are as follows: (1) description of the foundations; (2) applicable codes, standards, and specifications; (3) loads and load combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, and special construction techniques; (7) testing and inservice surveillance programs; (8) ITAAC; and (9) COL action items and certification requirements and restrictions. Not all of these areas were applicable to the review of the proposed changes to AP1000 DCD Section 3.8.5 and the following SER sections provide the staff's evaluation for the relevant areas.

In its previous evaluations of AP1000 DCD, Section 3.8.5, the staff identified acceptance criteria based on the design meeting relevant requirements in 10 CFR 50.55a, "Codes and Standards"; 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records"; GDC 2, "Design Bases for Protection Against Natural Phenomena"; and GDC 4, "Environmental and Dynamic Effects Design Bases." The staff found that the design of the AP1000 foundations was in compliance with these requirements, as referenced in NUREG-0800 Section 3.8.5 and determined that the design of the AP1000 foundations, as documented in the AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

In the AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.5 of the certified design:

1. As a result of the extension of the AP1000 design from just hard rock sites to sites ranging from soft soils to hard rock, various seismic re-analyses of the NI structures were performed. Whereas the original design relied upon the equivalent static method of analysis for seismic loading, the re-analyses included the additional use of response spectrum and time history methods of analysis. Appendix G of DCD Revision 17 indicates that the response spectrum analysis was used for the 3D refined finite element model of the NI and for the analysis of the PCS valve room and miscellaneous steel-framed structures, flexible walls, and floors. Time history analyses were used to determine maximum soil bearing pressures under the NI and, subsequent to the submittal of DCD Revision 17, to perform an updated NI stability evaluation.

2. In DCD Revision 16, the applicant revised Section 3.8.5.4.1 - Analyses for Loads during Operation, regarding the reinforcing steel under the shield building and the auxiliary building. Additional reinforcement is provided in the design of the basemat for soil sites such that the basemat can resist loads 20 percent greater than the demand calculated using the equivalent static acceleration analyses on uniform soil springs. The design accommodates potential site specific soil variability beneath the basemat in the horizontal (lateral) directions.

3. In DCD Revision 16, the applicant included in Section 3.8.5.4.2 a description of the analyses which evaluate the effects of different construction sequences on settlement and the design of the basemat. DCD Revision 17 made some additional revisions to describe the concrete placement sequence in the basemat and in the auxiliary building during construction.

4. In DCD Revision 16, the applicant revised Section 3.8.5.4.3 - Design Summary Report. DCD Revision 15 indicated that the results of the evaluation will be documented in an as-built summary report by the COL applicant. In DCD Revision 16, this was revised to state, "The results of the evaluation will be documented in an as-built summary report."

5. In DCD Revision 16, the applicant revised Section 3.8.5.4.4 - Design Summary of Critical Sections. The design approach of the basemat for two of the critical sections was revised to design these sections as two way slabs.

6. In DCD Revisions 16 and 17, several revisions were made in Section 3.8.5.5 - Structural Criteria, regarding the sliding and overturning stability evaluations. In DCD Revision 16, Section 3.8.5.5.3 - Sliding, the sliding-coefficient of friction between the basemat and the soil was revised from 0.55 to 0.70. In DCD Revision 17, Section 3.8.5.5.4 - Overturning, the equation used to calculate the factor of safety for overturning due to the safe shutdown earthquake was revised.

7. In DCD Revision 16, the applicant revised Section 3.8.5.6 - Materials, Quality Control, and Special Construction Techniques. DCD Revision 15 indicated that

the COL applicant would provide information related to the excavation, backfill, and mudmat. In DCD Revision 16, this was revised to state that Subsection 2.5.4.5.3 describes the information related to the excavation, backfill, and mudmat.

8. In DCD Revision 16, the applicant revised Section 3.8.5.7 - In-Service Testing and Inspection Requirements. DCD Revision 15 indicated that the COL applicant has the responsibility to determine the need for foundation settlement monitoring. In DCD Revision 16, this was revised to state that the need for foundation settlement monitoring is site-specific as discussed in subsection 2.5.4.5.10.

The evaluation of changes to the description of foundations, applicable codes, standards, and specifications, loads and load combinations, and the design and analysis procedures may be found in the evaluation of TR-85, presented below.

3.8.5.1 Nuclear Island Basemat Technical Report TR-85

Since the AP1000 design was previously certified for use at an HR site, the applicant submitted APP-GW-GLR-044, TR-85, "Nuclear Island Basemat and Foundation," Revision 0, to summarize the design of the NI basemat and exterior walls below grade for both HR and soil sites. This report also describes interface demands to be satisfied at a site. TR-85 Revision 0 indicates that the report also provides an updated baseline for the as-designed configuration and validates the basemat and foundation design against the updated seismic spectra and soil foundation conditions. TR-85 was subsequently modified in Revision 1 to address a number of the outstanding RAIs. Some of the information in TR-85 is included in the AP1000 DCD, Revision 17.

As a result of the staff's review of TR-85, a number of RAIs were sent to the applicant. Based on these RAIs, the applicant made a number of revisions in the analyses and design methods to address the issues raised. The description provided below presents the staff's evaluation of the key issues.

3.8.5.1.1 Design of NI Walls below Grade

As a result of the staff's review of TR-85, a number of questions were identified related to the design of the foundation walls below grade. These questions were captured in RAI-TR85-SEB1-02, RAI-TR85-SEB1-04, RAI-TR85-SEB1-34, and RAI-TR85-SEB1-40. As a result of these RAIs, the applicant made a number of revisions in the analyses and design methods to address the issues raised. The description provided below presents the staff's evaluation of the key issues related to the design of the foundation walls below grade.

As described in the applicant's response to RAI-TR85-SEB1-02, the analytical approach to calculate the pressure loads on the side walls below grade (embedded walls) consisted of hydrostatic pressure from ground water, at rest earth pressure, surcharge pressure, dynamic earth pressure, and passive earth pressure. The seismic earth pressure was calculated in accordance with ASCE 4-98, Section 3.5.3, which utilizes the elastic solution for dynamic soil pressures. In addition to designing the foundation walls to the seismic earth pressure, the RAI

response also indicates that the NI exterior walls are designed for the passive soil pressure in the load combinations that include SSE.

The staff finds that the approaches used by the applicant to calculate these various soil pressure loads were in accordance with industry-wide soil mechanics methods and were consistent with the criteria presented in NUREG-0800 Section 3.7 for seismic loads and Section 3.8 for design methods, and, therefore, are acceptable.

3.8.5.1.2 Maximum Soil Bearing Pressure beneath the Basemat during SSE

As a result of the staff's review of TR-85, a number of questions were identified related to the calculation of the maximum soil bearing pressures beneath the basemat due to the SSE. These questions related to soil bearing pressure were captured in RAI-TR85-SEB1-03, RAI-TR85-SEB1-04, RAI-TR85-SEB1-06, RAI-TR85-SEB1-15, RAI-TR85-SEB1-26, and RAI-TR85-SEB1-40. As a result of these RAIs, the applicant made a number of revisions in the analyses and design methods to address these issues. The description provided below presents the staff's evaluation of the key issues related to the soil bearing pressure evaluations.

Based on the response provided to RAI-TR85-SEB1-03, the maximum dynamic bearing pressure on soils resulting from SSE was 5745.6 kPa (120,000 pounds per square foot (psf)) for the HR case in the previous AP1000 certified design using the more conservative equivalent static analysis method. The 5745.6 kPa (120,000 psf) pressure was reduced to 1316.7 kPa (27,800 psf) for the HR case by using a more realistic 2D [ ] nonlinear (liftoff) analysis. The 2D [ ] nonlinear (liftoff) analysis showed that the SM soil case gives a somewhat higher dynamic bearing pressure, 1651.9 kPa (34,500 psf), than that of the HR case. The applicant also calculated the maximum dynamic bearing pressure on soils by using the [ ] 3D finite element NI20 model with a seismic time history SSI analysis. This analysis was performed for the HR case and five soil conditions, and the resulting maximum dynamic bearing pressure is 1675.8 kPa (35,000 psf). This analysis is described in detail in Section 2.4.3 of TR-85, Revision 1, and TR-03 (November 2008). The maximum soil bearing pressure demand of 1675.8 kPa (35,000 psf) for the NI is presented in AP1000 DCD Tier 1, Section 5.0, "Site Parameters." The applicant also explained how the time history analyses removed a number of conservatisms inherent in the equivalent static seismic analysis, which led to the large reduction in the soil bearing pressure. Based on this explanation and the use of a more accurate [ ] 3D finite element NI20 model analysis, which was also confirmed with the independent 2D nonlinear liftoff [ ] analysis, the staff concludes that the applicant has used proper methods to obtain the maximum dynamic bearing pressure on the soil.

3.8.5.1.3 Stability Analysis (Sliding and Overturning) of the Basemat and Foundation Waterproofing Systems

As a result of the staff's review of TR-85, a number of questions were identified related to the calculation of the stability analysis of the NI basemat and the foundation waterproofing systems. These questions were captured in RAI-TR85-SEB1-04, RAI-TR85-SEB1-07, RAI-TR85-SEB1-10, RAI-TR85-SEB1-11, RAI-TR85-SEB1-34, RAI-TR85-SEB1-35, and RAI-TR85-SEB1-40. As a result of these RAIs, the applicant made a number of revisions in the analyses and design methods to address these issues. The description provided below presents the staff's evaluation of the key issues related to the stability evaluations.

Based on the response to RAI-TR85-SEB1-10, for the overturning and sliding stability evaluation, the applicant initially used the 3D [ ]NI20 model. For the SSE loading, an equivalent static analysis was performed and demonstrated that without the use of passive soil pressure resistance, the overturning factors of safety were met. However, for sliding, difficulties were identified in satisfying the sliding factor of safety. Therefore, the applicant performed another more realistic nonlinear analysis with sliding friction elements using a modified 2D [ ] model that was used previously to study the basemat uplift. This model, which is described in Section 2.4.2 of TR-85, was modified to use sliding friction elements at the interface of the basemat and the soil. The model considered basemat vertical uplift in addition to sliding. A direct integration time history analysis using the modified 2D [ ] model was performed to evaluate the basemat stability issue. Three soil cases that have the lowest factor of safety-related to sliding were evaluated. These three cases are HR soil, UBSM soil, and SM soil. The seismic input was increased by 10 percent so as to maintain the factor of safety against sliding of 1.1. No passive soil resistance was considered in the analyses. The resulting maximum deflection at the base using a coefficient of friction of 0.55 was 0.08 cm (0.03 in) for all three soil cases. This horizontal sliding deflection was considered to be negligible and no passive soil pressure resistance was necessary from the backfill. Therefore, the applicant concluded that the NI is stable against sliding and there is no passive pressure required to maintain stability. The AP1000 DCD requires COL applicants to demonstrate by testing that soils beneath their basemat possess a minimum coefficient of friction of 0.7, which is equivalent to the soil friction angle of 35 degrees, and this provides additional conservatism for the basemat against sliding stability.

The staff's review of the RAI-TR85-SEB1-10 response related to the seismic stability evaluation of the NI concludes that the overall 2D [ ] nonlinear sliding analysis approach appears to be appropriate; nevertheless, a review of the applicant's calculation was needed to confirm the proper implementation of this methodology is appropriate. At the seismic audit conducted during the week of June 14, 2010, the staff reviewed the 2D [ ] non-linear sliding stability evaluation. As a result of this review a change was made to the [ ] sliding/contact finite element that resulted in larger horizontal displacements. The resulting maximum displacement at the base of the NI basemat was 0.14" without buoyant force consideration, and 0.24" with buoyant force effects. These values are larger than the previously reported results, 0.03" without buoyant force and 0.045" with buoyant force effects. However, these values are still judged to be negligibly small, especially when the conservative analysis approach of neglecting sliding resistance from the soil passage pressure and neglecting the additional fictional forces along the barrier portions of the NI side walls are considered. Therefore, it is concluded that the NI is stable against sliding. However, the staff notes the need to revise the response to RAI-TR85-SEB1-10 to reflect the revised finite element for sliding and the increase in displacements, and provide the DCD and TR-85 changes to reflect the sliding evaluation.

Since wind and tornados generate less horizontal sliding force and overturning bending moment than the SSE does, the applicant concluded that the NI, which does not have stability problems against SSE, will not have problems against wind and tornados.

As a result of the staff's structural audit conducted during the week of August 10, 2009, the NRC staff requested justification as to why TR-85 is not identified as Tier 2* since it is referenced in AP1000 DCD Section 3.8.5 and it includes key details of the design of the foundation. Similarly,

justification was not provided for identification of Tier 2* for TR-09 (Containment Vessel Design Adjacent to Large Penetrations), TR-57 (Nuclear Island: Evaluation of Critical Sections), and the updated shield building reports. Therefore, in a follow-up to RAI-TR85-SEB1-10, the staff requested that TR-09, TR-57, and TR-85 be identified as Tier 2* information in the AP1000 DCD, or an acceptable justification be provided.

At the seismic audit conducted during the week of June 14, 2010, the staff reviewed the 2D [ ] nonlinear sliding stability evaluation. As a result of this review, a change was made to the [ ] sliding/contact finite element, which resulted in larger horizontal displacements. The resulting maximum displacements at the base of the NI basemat were determined to be 0.36 cm (0.14 in) without buoyant force consideration, and 0.51 cm (0.24 in) with buoyant force effects considered. These values are larger than the previously reported results of 0.08 cm (0.03 in) without buoyant force consideration, and 0.11 cm (0.045 in) with buoyant force effects. However, these values are still judged to be negligibly small, especially when the conservative analysis approach of neglecting any sliding resistance from the soil passive pressure and neglecting the additional frictional forces along the buried portions of the NI side walls are considered. Therefore, it can be concluded that the NI is stable against sliding. However, the applicant must revise the response to RAI-TR85-SEB1-10 to reflect the revised finite element for sliding and the increase in displacements, and provide the mark-ups for the AP1000 DCD changes and TR-85 to reflect the changes in the sliding evaluation.

In response to the above requests, the applicant's letters dated July 30, 2010, and August 25, 2010, indicated that the applicant would review the information in the RAI responses and the structural TRs for the key analysis and design information that should be included in the AP1000 DCD, and would provide DCD mark-ups for the complete Sections 3.7 and 3.8, as well as Appendixes 3G, 3H and 3I, identifying the Tier 2* information. In addition, the applicant provided the mark-ups for the AP1000 DCD and TR-85 to reflect the changes in the sliding evaluation due to modifications for the sliding/contact finite element. The staff's review of the RAI responses in the two letters concluded that the proposed approach, to add the specific Tier 2* information from the applicable TRs and shield building report(s) to the AP1000 DCD, is acceptable because mark-ups will be provided and give the staff an opportunity to confirm that the required information will be identified as Tier 2* in the AP1000 DCD. The response regarding the revised NI seismic sliding evaluation is also acceptable because it provides the mark-ups for the changes to the AP1000 DCD and TR-85 to reflect the changes in the sliding evaluation and the increases in seismic displacement due to sliding. Because the identification of the specific Tier 2* information has not been completed and the corresponding AP1000 DCD mark-ups have not been provided, this issue is identified as Confirmatory Item CI-TR85-SEB1-10, which will be tracked and evaluated as part of the staff's confirmatory review of the AP1000 DCD. The staff notes that the applicant clarified the design basis by letters dated October 21, 2010, whereby they withdrew TR-57 and provided mark-ups of the DCD to show the removal of references to TR-57 and stated the location where the information, as updated, appears in the proposed DCD and an appendix thereto.

A concrete mud mat consisting of an upper and a lower mud mat is placed on top of the soil foundations to provide a level support for the structural concrete basemat. A waterproofing membrane is placed between the upper mud mat and the lower mud mat. In RAI-TR85-SEB1-35, the staff requested that the applicant describe, in greater detail, the types of waterproofing materials to be used and how the coefficient of friction for these materials,

assumed in the sliding stability evaluations, will be demonstrated. In response, the applicant explained that one of three types of waterproofing systems is used: plasticized polyvinyl chloride (PVC) membrane, high-density polyethylene (HDPE) membrane, or a crystalline spray type material. The AP1000 DCD requires COL applicants to demonstrate by testing that the waterproofing membrane will achieve a minimum coefficient of friction of 0.55 (the value which was used for the NI sliding stability analysis) between it and the concrete mud mat.

The staff's review of the applicant's responses to RAI-TR85-SEB1-35 determined that the information provided to describe the waterproofing materials was not sufficient and that further revisions in the AP1000 DCD were required to reflect the revised details of the waterproofing materials. The remaining items that needed to be addressed relate to the proposed mark-up in the AP1000 DCD describing the waterproofing materials, more detailed information about the type and industry standards used for the waterproofing membrane, and information that demonstrates the adequacy of the crystalline waterproofing material.

In the applicant's letter dated June 30, 2010, the response to RAI-TR85-SEB1-35 indicated that the waterproofing system for the below grade walls and mud mat would consist of either the HDPE double-sided textured membrane; HDPE single-sided adhering sheet membrane; self-adhesive, rubberized asphalt/polyethylene membrane (for walls only); or sprayed-on waterproofing membrane based on polymer-modified asphalt or polyurea. The response explained that the use of the crystalline waterproofing material had been eliminated as an option. In addition, the industry standards used to specify performance requirements and other design requirements (e.g., maximum crack width) for the waterproofing systems were provided. The proposed mark-ups to the AP1000 DCD describing the waterproofing materials and performance requirements were also provided and found to be acceptable based on the use of the applicable industry standards and industry practices. Also, the elimination of the use of the crystalline material resolves the questions raised regarding the adequacy of this material. Therefore, this is identified as Confirmatory Item CI-TR85-SEB1-35, pending revision of the AP1000 DCD.

3.8.5.1.4 The Effect of Basemat Liftoff from the Ground

Section 2.4.2 of TR-85, Revision 1, and the response to RAI-TR85-SEB1-14 described analyses performed using a 2D [ ] nonlinear model to evaluate the potential effects of liftoff. This was needed because [ ] analyses cannot model nonlinear behavior, such as liftoff of the NI structure from the soil. The [ ] analyses permit tension to be transferred across the interface between the basemat and the soil. Therefore, analyses were performed with the 2D [ ] nonlinear model, which allowed for liftoff, and the results were compared to 2D [ ] analyses, which do not have liftoff. The NI superstructures (i.e., structures above the basemat) were represented as stick models in both the 2D [ ] model and the 2D [ ] model. In the 2D [ ] model, the soil was represented by horizontal and vertical springs. The springs were only effective when the basemat was in contact with the soil (i.e., when the springs were in compression).

The results of the two analyses were compared in terms of FRS in the structures, member forces, and soil bearing pressures. The applicant provided comparisons of in-structure FRS, member forces and soil bearing pressures. The applicant indicated that these comparisons show that there is no significant difference between the 2D [ ] nonlinear analyses and the

2D [ ] linear analyses. On this basis the applicant concluded that the NI superstructure may be designed neglecting liftoff, but the basemat design does need to consider the effects of liftoff. Thus, Section 2.6 in TR-85 provides the analysis and design of the NI basemat, which uses a 3D [ ] model that does consider liftoff.

The staff review of the tabulated comparisons of the member forces at representative locations between the 2D [ ] and the 2D [ ] analyses showed a maximum difference of 2.7 percent. The in-structure generated response spectra comparisons at key locations showed that the 2D [ ] nonlinear analysis spectra were often below or within about 10 percent above the 2D [ ] linear results, except at the very low frequency of about 4.8 Hz in the vertical direction where the difference is about 15 percent. For soil bearing comparisons, the differences for the maximum soil bearing pressures were within about 6 percent. Since the applicant performed a nonlinear [ ] analysis with liftoff capability and showed that the results are reasonably close to the [ ] results without liftoff capability, the staff finds the applicant's approach for addressing the NI liftoff effects acceptable. Therefore, RAI-TR85-SEB1-14 is resolved.

3.8.5.1.5 Basemat Design

3.8.5.1.5.1 Seismic Analysis of NI Basemat and Soil Reaction Force (Pressure) at the Bottom of the Basemat

The seismic analysis was based on the 3D [ ] finite element NI05 model using seismic equivalent static accelerations, which were obtained from the time history analysis of the NI on HR, prior to the design changes made to enhance the shield building. This 3D [ ] NI05 analysis of the basemat is described in Section 2.6.1 of TR-85, Revision 1, and in the responses to RAI-TR85-SEB1-21, RAI-TR85-SEB1-22, and RAI-TR85-SEB1-23. The model is nonlinear because soil springs can only take compression but not tension when the basemat lifts off the ground. To verify the adequacy of the equivalent static accelerations used in the 3D [ ] NI05 model another study was performed. First, a linear analysis using the equivalent static accelerations discussed above was performed to determine the total base reactions and soil bearing pressures. Then, a time history fixed base analysis, which accounted for the various soil profiles, was performed. The time history inputs for this analysis were developed based on the envelope of the basemat responses given by the 3D [ ] analyses. The 3D [ ] analyses considered five soil cases: FR, SR, UBSM, SM, and SS. Based on the comparison of the base reactions and soil bearing pressures from the equivalent static analysis (for the HR condition) and the time history analysis (for the range of soil conditions), the applicant concluded that the study demonstrated that the equivalent static accelerations from the prior time history analysis of the NI on HR, are still acceptable.

The staff finds that the 3D [ ] NI05 model is appropriate since it was developed in accordance with industry methods and is consistent with the guidance presented in NUREG-0800 Section 3.8.5. The applicant's use of the equivalent static analysis as described above is reasonable because the applicant compared the base reactions from the NI and soil bearing pressures obtained from the equivalent static analysis with the results from the time history analysis that considered the range of possible soil conditions.

The soil pressure imposed on the bottom of the basemat, obtained from the above seismic analyses, is based on the assumption that the NI rests on a uniform soil site. For a site to be considered uniform, the variation of Vs in the material below the foundation to a depth of 36.7 cm (120 ft) below the finished grade within the NI footprint shall meet the criteria as stated in AP1000 DCD Section 2.5.4.5.3.

A 20 percent margin was provided in the design of the basemat, which was intended to account for possible soil property variations beneath the basemat at a site that may not meet the criteria for uniform soil sites. Additional analyses would be required for nonuniform soil sites. If the soil variations exceed the criteria as defined in AP1000 DCD Section 2.5.4.5.3, then the AP1000 DCD requires that an evaluation for nonuniform soil conditions be performed and this evaluation needs to be provided as part of the COL application. A procedure for evaluating the site-specific nonuniform soil condition is also provided in AP1000 DCD Section 2.5.4.5.3.

3.8.5.1.5.2 Soil Subgrade Modulus

In RAI-TR85-SEB1-05, the staff requested that the applicant provide a complete set of soil subgrade modulus values used for the AP1000 rock and soil cases. In a letter dated March 31, 2008, the applicant provided its response as follows:

- Subgrade moduli of 984.5, 502.7, 157.1, and 300.2 megapascal per cubic meter (MPa/m^3) (6267, 3200, 1000, and 300 kips per cubic feet (kcf)) were used for HR, SR, SM and SS sites in the 2D [ ] parametric linear dynamic analyses described in Section 2.4.2 of TR-85. The results of the analyses for SR and SS were not used.

- Subgrade moduli of 984.5 MPa/m^3 (6267 kcf) and 157.1 MPa/m^3 (1000 kcf) were used for the HR and SM soil sites in the 2D [ ] nonlinear dynamic analyses described in Section 2.4.2 of TR-85.

- A subgrade modulus of 984.5 MPa/m^3 (6267 kcf) was used for HR in the 3D [ ] equivalent static nonlinear analysis for design of the basemat as described in Section 2.3.1 of TR-85.

- A subgrade modulus of 81.7 MPa/m^3 (520 kcf) was used for soil sites in the 3D [ ] equivalent static nonlinear analysis for design of the basemat as described in Section 2.6.1 of TR-85.

- A subgrade modulus of 40.8 MPa/m^3 (260 kcf) was used in the 3D [ ] equivalent static nonlinear parametric analysis for evaluation of the effect of a lower subgrade modulus as described in Section 2.7.1.1 of TR-85.

TR-85, Revision 1, indicates that the design of the NI basemat is based on the soil subgrade modulus corresponding to 81.7 MPa/m^3 (520 kcf) (comparable to the SM soil condition). This value of soil subgrade modulus was determined to be the governing soil case for design of the basemat considering the range of soil properties from HR to SS. To address soil conditions potentially softer than 81.7 MPa/m^3 (520 kcf), a study was performed to evaluate the effects of using lower stiffness values for the soil. Based on the applicant's March 31, 2008, and January 9, 2009, letters, the staff identified a number of items that still needed to be addressed

regarding the evaluation for the appropriate range of subgrade modulus values. One of the concerns was that at other similar soil sites, subgrade modulus values as low as 6.3 MPa/m^3 (40 kcf) (static case) and about 12.6 MPa/m^3 (80 kcf) (dynamic case) have been identified. Therefore, in a follow-up to RAI-TR85-SEB1-05, the applicant was requested to explain whether the use of such low values had been considered and, if not, to provide the technical basis for not considering these values.

In a letter dated August 4, 2009, the applicant described the results of a study that was performed for a low soil modulus value of 12.6 MPa/m^3 (80 kcf) whose results were compared to the analysis using 81.7 MPa/m^3 (520 kcf) and 40.8 MPa/m^3 (260 kcf) soil moduli. To address the concern related to the design of the foundation, the RAI response indicates that a comparison of the 2D [ ] analysis results for all soil cases (FR, SR, UBSM, SM, and SS) was made to the soil case corresponding to a subgrade modulus of 12.6 MPa/m^3 (80 kcf). The results show that the soil bearing pressures for the 12.6 MPa/m^3 (80 kcf) soil case are very close to the 40.8 MPa/m^3 (260 kcf) (SS) case and they are bounded by the results for the 81.7 MPa/m^3 (520 kcf) case, which was used in the design of the basemat. The bending moments for the shield building at the base using the 81.7 MPa/m^3 (520 kcf) soil case bound the moments for the 12.6 MPa/m^3 (80 kcf) soil case. Therefore, the applicant concluded that these results demonstrate that the design of the foundation using a soil modulus value of 81.7 MPa/m^3 (520 kcf) is valid for soil subgrade moduli as low as 12.6 MPa/m^3 (80 kcf). For the soil bearing pressure demand, the comparisons presented in the RAI response show that the soil bearing pressure demand, used as interface criterion in the AP1000 DCD Tier 1, is acceptable since it bounds the soil bearing pressure for the 12.6 MPa/m^3 (80 kcf) case.

The staff found that the 2D [ ] analysis results demonstrate that the building responses for the 12.6 MPa/m^3 (80 kcf) soil modulus are bounded by the results for the 81.7 MPa/m^3 (520 kcf) soil case, which was used for design of the structures and for determining the soil bearing pressure demand. Also, for stability evaluation, the results presented in TR-85, Revision 1, show that the seismic shear force and overturning moment are lower when softer soil conditions are considered. Therefore, the stability evaluation performed by the applicant would also bound the results obtained with a reduced soil modulus of 12.6 MPa/m^3 (80 kcf). Based on the above discussion, the staff concludes that the soil cases used by the applicant for design, soil bearing pressure demand, and stability evaluation address the staff's concerns regarding subgrade moduli values lower than 81.7 MPa/m^3 (520 kcf). Therefore, RAI-TR85-SEB1-05 is resolved.

3.8.5.1.5.3 Assumption of Uniform Soil Pressure beneath the Basemat

The applicant assumed uniform soil pressure acting on the bottom of the basemat in its analysis for bending moments and shear forces in the basemat. It is a well known phenomenon in soil mechanics that the soil pressure is higher at the edge of the basemat than it is away from the edge, which is referred to as the Boussinesq effect. Therefore, in RAI-TR85-SEB1-32, the staff requested that the applicant demonstrate that the use of the uniform soil springs for the design of the basemat is justifiable, where the actual distribution of the soil stiffness would not be uniform.

The RAI responses, dated June 23, 2009, and October 19, 2009, presented the results of a study, that compared soil bearing pressures due to dead load at the bottom of the basemat from the uniform soil springs and the finite element representation of the soil. However, these results

showed that the soil bearing pressure along the horizontal interface between the basemat and the soil do not appear to compare well in some regions. Furthermore, separate moment contour plots were provided for the basemat corresponding to each soil stiffness representation; however, without a direct quantitative comparison of member forces it is difficult to judge that the use of the uniform soil springs for the design of the foundation is acceptable. In a follow-up RAI, the staff requested that the applicant clearly demonstrate that the bending moments and shear forces in the basemat using uniform soil springs are acceptable by providing quantitative data from the study at locations in the basemat that govern the design.

Based on the applicant's letter dated June 19, 2010, a study was performed to compare the uniform soil spring approach with the more accurate finite element soil representation that is able to capture the Boussinesq effect in soils. This study showed that the soil pressures are not uniform and that some member forces in the critical sections in the basemat were larger using the finite element soil model. The applicant tried to scale the prior design results to show that the design is still adequate for the increased loads. However, the response to the RAI did not adequately demonstrate that the design met the code limits.

In a letter dated July 30, 2010, the applicant provided the re-evaluation for the basemat design using the increased loads from the finite element model for the critical (governing) sections and using the permissible redistribution of moments in accordance with the ACI-349 Code. In addition, the applicant provided the results for the various 100-40-40 seismic combination methods used for the design of the basemat. The staff's review of the response determined that several items still needed to be addressed, primarily because the response to the RAI still did not adequately demonstrate that the design met the code limits. Nor was the use of the WEC 100-40-40 method appropriate. Therefore, in a follow-up RAI, the staff requested that the applicant justify the use of the 20 percent moment redistribution; show that the reinforcement design meets code requirements; provide the comparison for the WEC 100-40-40 method versus the ASCE 4-98 industry method; and demonstrate that there are no significant increases in the basemat forces due to potential concrete cracking.

In response to the above requests, the applicant's letter dated September 8, 2010, provided detailed information justifying the use of the 20 percent moment redistribution in accordance with the ACI-349 Code. In addition, according to the letter, a new study was performed to compare the results from a 2D nonlinear (with lift-off capability) equivalent static analysis using the WEC 100-40-40 method with those from a 2D nonlinear (with lift-off) time history analysis. The study shows that the maximum basemat bearing pressure from the 2D static analysis with the WEC 100-40-40 method in two dimensions is about 30 percent higher (i.e., more conservative) than that of the bearing pressure from the more accurate 2D dynamic time history analysis approach. To address the effect of concrete cracking on the basemat forces, the applicant performed another study, which provided a comparison of the FRS at representative locations in the NI, which shows that the ZPAs obtained from the nonlinear analysis (that considers cracking of concrete) were reasonably close to the ZPAs obtained from the linear analysis using a stiffness reduction factor of 0.80, which was assumed in the design basis analysis.

The staff review of the response concluded that: (1) the justification for the use of 20 percent moment redistribution is acceptable because the information provided demonstrates that the provisions in ACI-349 regarding negative moment redistribution have been satisfied; (2) the

basemat design based on the 2D nonlinear (with lift-off) equivalent static analysis using the WEC 100-40-40 method is conservative based on the applicant's study comparing the results to the more accurate 2D nonlinear time history analysis, which inherently includes the phasing of the different input components; and (3) there is no significant increase in the basemat forces due to concrete cracking in the NI, because another study was performed to demonstrate that the use of the 0.8 stiffness reduction factor adequately accounts for cracking.

Based on the above discussion, this is being tracked as Confirmatory Item CI-TR85-SEB1-32, pending revision of the AP1000 DCD and TR-85 to incorporate the mark-ups proposed in the RAI response.

3.8.5.1.5.4 Load Combinations and Reinforcement Design

As a result of the staff's review of TR-85, a number of questions were identified related to the load combinations and design of the basemat reinforcement. These questions were captured in RAI numbers TR85-SEB1-28, TR85-SEB1-29, and TR85-SEB1-30. As a result of these RAIs, the applicant made a number of revisions in the analyses and design methods to address these issues. The description provided below presents the staff's evaluation of the key issues related to the load combinations and design of the basemat reinforcement.

In RAI-TR85-SEB1-28, the staff requested that the applicant explain why the load combinations presented in the TR-85 were not consistent with those in Table 3.8.4-2 of the AP1000 DCD. In a letter dated December 2, 2008, the applicant provided a mark-up of AP1000 DCD Table 3.8.4-2 to be consistent with the revised TR-85. The staff finds that the new load combinations in the mark-up of AP1000 DCD Table 3.8.4-2 and in the revised TR-85 are in accordance with the ACI-349-01 Code, and, thus, are acceptable. This is identified as Confirmatory Item CI-TR85-SEB1-28, pending revision of the AP1000 DCD.

In RAI-TR85-SEB1-29, the staff requested that the applicant describe the design approach used for the basemat in accordance with ACI-349-01. The staff also asked whether every 3D [ ] finite element is designed for the resultant forces in accordance with the ACI-349 Code and whether this process is automated by using a computer code or by hand calculations. In a letter dated October 19, 2007, the applicant stated that the design procedure is described in [ ], Revision 1, Section 4.2, "Calculation Approach/Methodology," and the calculation process is automated by a computer code. During the review of the shield building design, the staff found a potential error in the code. In the applicant's letter dated July 9, 2010, the response provided an explanation as to why some of the results from the computer code may have appeared as an error but they were not. The RAI response explained that the negative value of shear shown in the computer code results indicates that the code has detected that the concrete is in tension beyond its limit. The computer code does not use the strength provided by the concrete in that case. Based on the review of the design approach presented by the applicant for the basemat, the use of the ACI-349-01 Code for sizing the concrete sections and selection of reinforcement, and the information provided in the RAI response, which explained why negative values for shear may appear in the results generated from the computer code, the staff concludes that the design approach is acceptable. Therefore, RAI-TR85-SEB1-29 is resolved.

3.8.5.1.5.5 Minimum Required Soil Friction Angle, Settlement Criteria for the NI Structure, and Construction Sequence

Section 5.1 of TR-85 presents the proposed revisions to AP1000 DCD Tier 2, Table 2-1, which includes the site parameters including those for the soil media. Section 5.2 presents the proposed revisions to AP1000 DCD Tier 1, Table 5.0-1, which also includes the site parameters for the soil. Considering that the foundation of the AP1000 design has been extended to soil sites, in RAI-TR85-SEB1-36, the staff requested that the applicant include, in both tables, two additional parameters, which are needed for the structural design of the NI: a minimum required soil friction angle of 35 degrees beneath the basemat and settlement criteria for the NI structure.

In a letter dated March 31, 2008, the applicant provided the following response:

a) The minimum required soil friction angle of 35 degrees has been added to both Tables 2-1 and 5.0-1.

b) AP1000 DCD Section 2.5.4.6.11 requires the COL applicant to evaluate settlement at soil sites. The effect of settlement on the NI basemat during construction has been considered in the design of the NI as described in Section 2.5 of the report and in AP1000 DCD Section 3.8.5.4.2. These analyses considered the flexibility of the basemat during construction by performing a nonlinear analysis of the soil and NI. The nonlinear analyses are described in the applicant's response to RAI-TR85-SEB1-19, dated March 31, 2008. The analyses used the NI05 building model described in AP1000 DCD Appendix 3G. The analyses considered an SS site with properties selected to maximize the settlement during construction. Immediate settlements were based on elastic properties of the foundation medium, while the time-related settlements used creep parameters established by comparison against one-dimensional consolidation theory. These analyses show total settlements of about one foot.

The applicant has established guidance on settlement for the COL applicant in the RAI response. The acceptable criteria are as follows: Acceptable differential settlement between buildings without additional evaluation is identified as 7.6 cm (3 in) between the NI and the Turbine Building, the Annex Building, and the Radwaste Building. The 7.6 cm (3 in) is measured from the center of the Containment Building to the center of the Turbine Building, center of the Annex Building, or the center of the Radwaste Building. Each building, including the NI, also has a settlement criterion of no more than 1.3 cm (½ in) in 15.2 m (50 ft) in any direction. The NI also has an acceptable maximum absolute settlement value of 7.6 cm (3 in). If site-specific settlement analyses predict settlements below the values in this table, the site is acceptable without additional evaluation. If the analyses predict greater settlement, additional evaluation will be performed. This may include specification of the initial building elevations, specification of the stage of construction and settlement for making connections of systems between buildings, etc. It would also include review of the effect of the rotation of buildings and its effect on the gap between adjacent structures. These analyses would provide the basis for review of settlement measurements during construction and subsequent operation.

Regarding part a) of the RAI response, the staff noted that in a letter dated June 10, 2009, the applicant indicated that a soil internal friction angle of 35 degrees is required beneath the basemat and it is specified in Table 2-1 of the AP1000 DCD, and that the second paragraph of

AP1000 DCD Section 2.5.4.6.2 is revised to state that if the minimum soil angle of internal friction is below 35 degrees, the COL applicant will evaluate the seismic stability against sliding as described in Section 3.8.5.5.3 using the site-specific soil properties. The applicant also decided to remove the criterion for the soil friction angle of 35 degrees from the prior versions of AP1000 DCD Tier 1, Table 5.0-1, "Site Parameters." After reviewing the applicant's submittals, the staff requested that the applicant address several issues discussed below.

During the August 10, 2009 audit, the staff informed the applicant that if a site-specific evaluation is required for sliding because the soil friction angle is less than 35 degrees, then Section 3.8.5.5.4 of the AP1000 DCD should also add the evaluation requirement for overturning stability. In addition, the staff considered the demonstration of a site soil friction angle of 35 degrees to be a key site parameter in the stability evaluations and other analyses, such as determining the soil pressure loads for the design of the NI foundation walls. Therefore, this criterion should remain in AP1000 DCD Tier 1, Table 5.0-1. In a letter dated September 22, 2009, the applicant provided a proposed mark-up of AP1000 DCD Tier 1, Table 5.0-1, and AP1000 DCD Tier 2, Section 2.5.4.6.2, to incorporate the requirement for a site-specific evaluation when the soil friction angle is less than 35 degrees. However, the wording in AP1000 DCD Table 5.0-1, for the requirement of a site-specific evaluation, needs to be clarified so that it is clear that a stability evaluation should be performed for both sliding and overturning stability. In a letter dated May 14, 2010, the applicant revised the wording in the proposed mark-ups to AP1000 DCD Tier 1, Table 5.0-1, and AP1000 DCD Tier 2, Section 2.5.4.6.2. Since the AP1000 DCD markups specify the requirement for a minimum soil angle of internal friction of 35 degrees, and if it is less than 35 degrees, then the COL applicant will perform a site-specific analysis to demonstrate stability (sliding and overturning), the staff's review of the information concluded that the response is acceptable. Therefore, this is being tracked as Confirmatory Item CI-TR85-SEB1-37, pending revision of the AP1000 DCD to incorporate the mark-ups proposed in the RAI response.

Regarding part b) of the RAI response, the staff observed that if acceptable soil sites are already known to cause potential settlements of as much as one foot as previous studies have indicated, then the construction settlements will in fact exceed the listed limitation of 7.6 cm (3 in) for most soil sites. The staff requested that the applicant explain: (a) what should be the detailed plan that the COL applicant needs to implement when the predicted settlements in fact exceed 7.6 cm (3 in); and (b) if any of the predicted settlements are less than 7.6 cm (3 in) for the total settlement, as well as less than the other acceptance values presented in AP1000 DCD Table 2.5-1, while the actual measured settlements during construction are found to exceed these values before completion of construction, what is the impact on the ongoing construction process and what the COL applicant is supposed to do at that time.

In the applicant's letters dated December 2, 2008, and July 21, 2009, additional information was provided and one of the settlement threshold values was revised. The limit of acceptable settlement without additional evaluation was raised to 15.2 cm (6 in) for the total NI foundation mat. The RAI response also explained what steps would be taken in case the COL applicant's predicted settlement analysis for the site-specific conditions exceeds these limits.

The staff reviewed the information regarding the settlement criteria and concluded that the applicant has evaluated the effects of settlement on the structural integrity of the NI and that conservative settlement threshold values (i.e., lower than the settlement values used for

evaluation of the NI) have been proposed for inclusion in the AP1000 DCD. However, as requested in the original RAI and supplemental RAIs, the settlement criteria in the proposed mark-up of AP1000 DCD Tier 2, Table 2.5-1, should also be presented in AP1000 DCD Tier 1, Table 5.0-1.

In response to the above request, the applicant's letter dated June 21, 2010, indicated that the settlement criteria in the proposed mark-up of AP1000 DCD Tier 2, Table 2.5-1, are added to AP1000 DCD Tier 1, Table 5.0-1. Therefore, this is being tracked as Confirmatory Item CI-TR85-SEB1-36, pending revision of the AP1000 DCD to incorporate the mark-ups proposed in the RAI response.

In Section 2.5 of TR-85, the first paragraph states that in the expected basemat construction sequence, concrete for the mat is placed in a single placement. The last sentence of the same paragraph states that once the shield building and auxiliary building walls are completed to El. 25.1 m (82 ft-6 in), the load path changes and loads are resisted by the basemat stiffened by the shear walls. In RAI-TR85-SEB1-17, the staff requested that the applicant address several items related to the construction sequence. The applicant was requested to address issues related to the concrete pour of such a massive single concrete placement, how residual stresses at the junction between the shear walls and the shield building are calculated considering the construction sequence, and where in the AP1000 DCD the requirement to follow the construction sequences considered by the applicant in the design of the NI structures is located.

In a letter dated March 31, 2008, the applicant provided information to address the various items identified in the RAI. Regarding the construction sequence, the applicant described three construction sequences that were evaluated for an SS site to demonstrate construction flexibility within broad limits. The acceptability of the construction sequence used by the COL applicant is addressed by an ITAAC. The three construction sequences are as follows:

- A base construction sequence, which assumes no unscheduled delays.
- A delayed shield building case, which assumes a delay in the placement of concrete in the shield building while construction continues in the auxiliary building.
- A delayed auxiliary building case, which assumes a delay in the construction of the auxiliary building while concrete placement for the shield building continues.

The applicant indicated that analyses of alternate construction scenarios showed that member forces in the basemat are acceptable subject to the following limits imposed for SS sites on the relative level of construction of the buildings prior to completion of both buildings at El. 25.1 m (82 ft-6 in):

- Concrete may not be placed above El. 25.6 m (84 ft-0 in) for the shield building or CIS.
- Concrete may not be placed above El. 35.8 m (117 ft-6 in) in the auxiliary building, except in the CA20 structural module where it may be placed to El. 41.1 m (135 ft-3 in).

Based on the staff's evaluation of this response and follow-up RAI responses, the applicant was requested to revise the RAI response and Sections 2.5 and 3.8.5 of the AP1000 DCD to clearly state that in addition to satisfying settlement criteria the construction sequence limitations presented in Section 3.8.5.4.2 must be satisfied by the COL applicant. In the letter dated October 19, 2009, the applicant provided the proposed mark-up of AP1000 DCD Sections 2.5 and 3.8.5.4.2. The proposed wording indicates that the construction sequence limitations are only applicable to soil sites and not foundations identified by the applicant as SR, FR, or HR. The staff requested that the applicant justify why no construction sequence limitations are needed for the stiffer foundation materials.

In the applicant's letter dated July 15, 2010, the response to RAI-TR85-SEB1-17 indicated that the construction of the AP1000 will satisfy the construction sequence limits shown in AP1000 DCD Section 3.8.5.4.2 or a site-specific analysis of settlement and member forces will be completed. These limits do not apply to AP1000 units with a soil profile where Vs exceeds 2286.0 m/s (7500 fps). The Vs at the bottom of the basemat (i.e., locally) can drop to 2286.0 m/s (7,500 fps), while maintaining a Vs equal to or above 2438.4 m/s (8,000 fps) at the lower depths. The staff reviewed the proposed mark-ups to the AP1000 DCD and concluded that they are acceptable because: (1) the AP1000 was designed for the various construction sequences; and (2) the construction sequence limitations used in the SS evaluation are imposed on all soil conditions except for rock conditions having a Vs greater than 2286.0 m/s (7,500 fps). This is identified as Confirmatory Item CI-TR85-SEB1-17, pending revision of the AP1000 DCD.

3.8.5.1.5.6 The Effect of Ground Water on Nuclear Island Structures

The design of the AP1000 plant is based on saturated soil conditions. In RAI-TR85-SEB1-40, the staff requested that the applicant explain whether unsaturated conditions were also considered in performing any SSI analyses to determine the effects of unsaturated soils on the response of the NI in terms of member forces, deformations, and FRS.

In a letter dated May 27, 2009, the applicant indicated that it performed a time history analysis using a saturated and unsaturated SM soil profile (Poisson's ratio = 0.35) and compared the FRS of the two analyses. Generic SSI analyses for the AP1000 assume the water table to be at grade level with saturated soil properties supporting the NI. The unsaturated soil profile was produced from a SHAKE analysis where the water table was assumed to be well below the NI. The results of this analysis indicated that the depth of the water table used for SSI analyses has a negligible effect on the FRS at the key nodes. This study shows that generally the FRS for these two cases are very close to one another, with the spectra from saturated conditions somewhat higher in a few isolated cases. Since the FRS differences between the two models are negligible, no additional analyses are required to compare member forces or deformations.

The staff reviewed the applicant's submittal regarding the effect of saturated and unsaturated soil conditions on NI structures, and found the applicant's approach to address the issue reasonable and acceptable. Since the study shows that, generally, the FRS for both saturated and unsaturated cases are very close to each other, with the spectra from saturated conditions somewhat higher in a few isolated cases, and the design of the AP1000 plant is based on the saturated conditions, the staff concludes that the AP1000 design using saturated soil conditions adequate and acceptable. Therefore, RAI-TR85-SEB1-40 is resolved.

3.8.5.1.5.7 Potential Uplift/Sliding between CIS and Containment, and between Containment and Basemat

In RAI-TR85-SEB1-12, the staff requested that the applicant explain how the potential uplift and sliding between the CISs concrete base and the steel containment shell is addressed for the various soil conditions, and provide the basis for the statement in Section 3.8.2.1.2 of the AP1000 DCD, which indicates that the shear studs provided between the containment and concrete basemat below the containment are not required for design basis loads, but provide additional margin for earthquakes beyond the SSE.

In a letter dated October 19, 2007, the applicant stated that its analyses of stability for the HR site demonstrated that there was no uplift or sliding at the interface of the CIS and the CV. These analyses showed potential uplift of the CV and CISs from the NI basemat for the review level earthquake (RLE). Based on these analyses, the applicant provided shear studs between the CV and the NI basemat to provide additional margin for the RLE. These studs were then designed to accommodate pressurization of the CV. The number of studs required for containment pressure was more than double the number required for seismic overturning for the RLE at the HR site. Revision 1 of TR-85 describes the analysis, which demonstrated that no uplift or sliding occurs between the CIS and the containment, and between the containment and the basemat for both design basis SSE level of 0.3g and RLE level of 0.5g PGA for HR and all soil conditions. Based on this, RAI-TR85-SEB1-12 is resolved.

3.8.5.1.5.8 The 100-40-40 Method for Combining Three Components of Earthquake Motions

AP1000 DCD Section 3.7.2 states that the 100-40-40 method is used for combining the three components of earthquake motions for the NI basemat analyses, CV analyses and shield building roof analyses. NRC regulatory guidance in RG 1.92 and NUREG-0800 Section 3.7.2 indicates that the use of the 100-40-40 combination method is only acceptable for combining the co-directional responses, such as Mxx due to north-south, east-west, and vertical directions in order to obtain a combined Mxx. However, it does not appear from a review of TR-85 and AP1000 DCD Section 3.8 that the applicant has combined the three components in accordance with RG 1.92 and industry standard ASCE 4-98. This issue was also identified during the staff's evaluation of TR-57 and APP-1200-S3R-003 for the shield building, which is discussed in Section 3.8.4.1.1 of this SER. The issue of the proper implementation of the 100-40-40 method was captured under RAI-TR85-SEB1-27.

As indicated in a letter dated July 3, 2010, the applicant's approach for the 100-40-40 method (WEC 100-40-40 method) was used for both seismic linear and nonlinear equivalent static analyses for the design of the NI basemat, the SCV and the shield building roof. In addition, the applicant also indicated that: (1) for the basemat, the justification for using the applicant's 100-40-40 method was addressed under RAI-TR85-SEB1-32; (2) for the SCV, the adequacy of using the applicant's 100-40-40 method for the SSE loading condition was confirmed by a direct comparison of the combined seismic stress results against those from the more accurate time history analysis; and (3) for the shield building roof, a comparison of the applicant's 100-40-40 method to the ASCE 4-98 method was made. For the shield building roof analysis and design, the applicant developed equivalent static accelerations, such that the resulting member forces would envelope those from the RSA, performed for the input motion

applied at the foundation level enveloping all the soil cases. The justification for using the applicant's 100-40-40 method was provided by comparing the combined member forces corresponding to the 24 cases of the applicant's 100-40-40 method with the member forces from the ASCE 4-98 method.

The staff's review of the information provided to the staff concluded that: (1) the justification for using the applicant's 100-40-40 method under RAI-TR85-SEB1-32 is acceptable since this approach is coupled directly with the basemat design issue under RAI-TR85-SEB1-32, which was previously reviewed above; and (2) the response for the SCV is acceptable, because the results provided show that the applicant's 100-40-40 method produced conservative results when compared with the more accurate time history analysis results. However, the response for the shield building roof provided insufficient information, primarily because the comparison of the applicant's 100-40-40 method with the ASCE 4-98 method is only made for member forces and not the final design parameter (e.g., required reinforcement for concrete members or stress level for steel members). Therefore, it is not clear that the applicant's 100-40-40 method is adequate. To address the issue of the proper implementation of the 100-40-40 method for the shield building roof design, the staff requested the applicant to identify the locations where the 100-40-40 method was applied in the shield building roof design; determine the maximum required reinforcement (or stress levels for steel members) using the 24 cases of the applicant's 100-40-40 method (as is done in the applicant's design process) and compare these results with the required reinforcement (or stress levels for steel members) using the NRC-accepted SRSS method or the ASCE 4-98 100-40-40 method.

In response to the above requests, the applicant's letter dated September 23, 2010, identified that the air inlet, the tension ring and the composite radial steel beams were designed using the applicant's 100-40-40 method, and provided figures and descriptions of the models used for the design of the shield building roof. To justify the use of the applicant's 100-40-40 method, the applicant presented comparisons for the final design parameters for these members showing that, although in some cases the applicant's 100-40-40 method was nonconservative when compared with the SRSS method or the ASCE 4-98 method, in all cases the design of these members is still acceptable. This was demonstrated for concrete members by showing that the required reinforcement using the NRC-accepted SRSS method was less than the provided reinforcement and for steel members by showing that the calculated stresses using the NRC-accepted SRSS method were less than the code allowable.

Based on the above discussion, this is being tracked as Confirmatory Item CI-TR85-SEB1-27, pending revision of the AP1000 DCD and TR-85 to incorporate the mark-ups proposed in the RAI response.

3.8.5.1.6 Record Keeping Issues

Sections 2.3.1, 2.4.1, 2.4.2, and 2.6.1 of TR-85 indicate that equivalent static nonlinear analysis, 2D [ ] analysis, 2D [ ] linear dynamic analysis, 2D [ ] nonlinear time history analysis, 3D [ ] equivalent static nonlinear analysis, and others were performed. In RAI-TR85-SEB1-04, the staff requested that the applicant develop a table (or tables) similar to AP1000 DCD Tables 3.7.2-14 and 3.7.2-16 to show: (1) the purpose of each analysis; (2) the model type(s); (3) analysis method(s); (4) soil condition(s); (5) loads, load combinations,

combination method (for combining loads and directional combinations for SSE); (6) governing design loads; and (7) reference location in TR-85 or other reports for the detailed description.

In a letter dated December 4, 2007, the applicant provided revisions to the AP1000 DCD tables to show the additional information requested in this RAI and to reflect the changes in the methodology described in other RAI responses. Although sufficient information to describe the evaluations performed for the bearing pressure demand, foundation stability, and design of the basemat, has been provided in this and other RAI responses and in TR-85, Revision 1, the staff could not identify where a description of the evaluations for bearing pressure demand and foundation stability are presented in the AP1000 DCD. Therefore, the staff requested that the applicant include in the AP1000 DCD a description of the evaluations performed for the bearing pressure demand and foundation stability, which consists of a summary of the analyses presented in TR-85, Revision 1.

In a letter dated June 4, 2009, the applicant provided the proposed changes to the AP1000 DCD that describe in more detail the soil bearing pressure evaluation in TR-85. This information will be added to Appendix 3G of the AP1000 DCD. In addition, the applicant indicated that the changes to the AP1000 DCD related to the stability evaluation are given in a revision to RAI-TR85-SEB1-10, along with a summary of the 2D nonlinear sliding evaluation. Thus, the description of the stability evaluation for inclusion in the AP1000 DCD is evaluated separately under the staff's assessment of RAI-TR85-SEB1-10 in this SER. Based on the above, this is being tracked as Confirmatory Item CI-TR85-SEB1-04, pending revision of the AP1000 DCD.

TR-85 is referenced in AP1000 DCD Section 3.8.5 and it includes key analysis and design information of the foundation. TR-09 is referenced in AP1000 DCD Section 3.8.2.4.1 and it includes key analysis and design information for the containment. TR-57 is referenced in Revision 17 to the AP1000 DCD Section 3.8.4 and it includes key analysis and design information for the CIS, auxiliary, and the shield building critical sections. The staff notes that the applicant clarified the design basis by letters dated October 21, 2010, whereby they withdrew TR-57 and provided mark-ups of the DCD to show the removal of references to TR-57 and stated where the information, as updated, appears in the proposed DCD and an appendix thereto. APP-1200-S3R-003 is referenced in AP1000 DCD Section 3.8.4 and it describes key analysis and design information for the shield building. Any revisions to the Tier 2* information will be subject to the NRC review and approval to avoid unintended safety consequences. This is identified as Confirmatory Item CI-TR85-SEB1-10.

In RAI-TR85-SEB1-39, the staff requested that the applicant identify the specific design reports, calculations, and reports related to various studies that are applicable to the analysis and design of the AP1000 NI basemat and foundation.

In a letter dated October 19, 2007, the applicant provided the following response:

APP-1010-S3R-001, "AP1000 Design Summary Report: Nuclear Island Basemat," provides a detailed summary of the design of the NI basemat. It satisfies the guidelines of NUREG-0800 Section 3.8.4 and is available for review by the NRC staff during the structural audit.

The design summary report identifies the applicant's specific design reports, calculations, and reports applicable to the analysis and design of the AP1000 NI basemat and foundation. Some of the documents referenced therein are listed below. The criteria and methodology documents were previously reviewed during the audit of the basemat design on HR.

1. APP-GW-C1-001, "AP1000 Civil/Structural Design Criteria," Revision 1
2. APP-GW-S1-008, "Design Guide for Reinforcement in Walls and Floor Slabs," Revision 1
3. APP-GW-S1-009, "Design Guide for Thermal Effects on Concrete Structures," Revision 0
4. APP-1000-CCC-001, "Verification of Design Macro for Reinforced Concrete Walls and Floors," Revision 2
5. APP-1000-CCC-002, "Guidance on Checking Results of Design Macro Calculation," Revision 0
6. APP-1010-S2C-003, "Macro to Calculate Required Reinforcement in Solid Elements," Revision 0
7. APP-1010-S2C-004, "Basemat Liftoff, and CV Pressure Analyses for Nuclear Island with Soil," Revision 0
8. APP-1010-CCC-001, "AP1000 Basemat Design Report," Revision 2
9. APP-1010-CCC-003, "Basemat Design Studies, Effect of Soil Modeling," Revision 0
10. APP-1010-CCC-004, "Basemat Design, Below Auxiliary Building," Revision 1
11. APP-1010-CCC-005, "Basemat Design, Below Shield Building," Revision 0
12. APP-1200-S2C-002, "ASB Thermal and Earth Pressure Analyses," Revision 1
13. APP-1200-S2C-003, "Auxiliary Building Load Combinations and Loads for Finite Element Analyses," Revision 0
14. APP-1000-CCC-005, "N.I. - Design Loads, Exterior Walls Below Grade," Revision 1
15. APP-1000-CCC-004, "Nuclear Island Stability Evaluation," Revision 1
16. APP-1000-S2C-064, "Effects of Basemat Liftoff on Seismic Response," Revision 4
17. APP-1000-S2C-065, "Nuclear Island Stick Model Analyses at Soil Sites," Revision 0

In an e-mail dated April 30, 2009, the applicant updated the documents related to the basemat design that are available for review. In the audit conducted during the week of May 4, 2009, the

staff reviewed a number of these documents to ensure that the evaluations were performed in accordance with the AP1000 DCD and NRC regulatory guidance. The staff concluded that the applicant has identified the design reports, calculations, and reports related to the AP1000 NI basemat and foundation, and the staff had an opportunity to review some of these documents for technical adequacy. Therefore, RAI-TR85-SEB1-39 is resolved.

3.8.6 Combined License Information

Section 3.8.6, "Combined License Information" of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.6 of the certified design:

1. In DCD Revision 16, the applicant revised Section 3.8.6.1, Containment Vessel Design Adjacent to Large Penetrations. This revision eliminated this COL information item because the applicant indicated that the information had been addressed in APP-GW-GLR-005 (TR-09) and the applicable changes were incorporated into the DCD.

2. In DCD Revision 16, the applicant also revised Sections 3.8.6.2 through 3.8.6.4, to delete the remaining COL information items relating to the PCS water storage tank examination, as-built summary report, and in-service inspection of containment vessel. No explanation for this deletion was provided in DCD Section 3.8.

The staff evaluation of the changes to the COL information item in AP1000 DCD Section 3.8.6.1 related to the CV design adjacent to large penetrations is presented in Section 3.8.2.4.1 of this SER where the staff reviewed APP-GW-GLR-005, Revision 0 (TR-09). In RAI-SRP3.8.6-SEB1-01, the staff identified that the adequacy of the revision to AP1000 DCD Section 3.8.6.1 could not be determined until the staff completes its review of TR-09. Since that time, the review of TR-09 has been completed; the NRC staff has identified this as Confirmatory Item CI-SRP3.8.6-SEB1-01, pending revision of the AP1000 DCD and TR-09 to reflect the proposed mark-ups in the RAI responses related to TR-09.

The staff noted that the applicant removed the COL information items in AP1000 DCD Sections 3.8.6.2 through 3.8.6.4 that relate to the PCS water storage tank examination, as-built summary report, and the inservice inspection of containment vessel. Therefore, in RAI-SRP3.8.6-SEB1-01, the staff requested that the applicant restore these items in AP1000 DCD Section 3.8.6 which were discussed in the prior versions of AP1000 DCD Sections 3.8.1 through 3.8.5. In a letter dated February 19, 2009, the applicant indicated the following:

For the COL information item in AP1000 DCD, Section 3.8.6.2, the requirement to examine the PCCWST is redundant with Design Commitment 10, ITAAC Item ii of Tier 1, Table 3.3-6.

For the COL information item in AP1000 DCD Section 3.8.6.3 the requirement to prepare an as-built summary report is redundant with Design Commitment 2.a, ITAAC Item I of Tier 1 Table 3.3-6.

For the COL information item in AP1000 DCD Section 3.8.6.4, the inservice inspection of the containment is required by NRC regulations including 10 CFR 50.55a. There is also a commitment for inservice inspection of the containment in AP1000 DCD Section 6.6.1.

The staff's review of the information provided in the RAI response has led to the conclusion that the deletion of the COL information item in AP1000 DCD Section 3.8.6.3 is acceptable because the information is redundant with an ITAAC and, in the case of Section 3.8.6.4, is already required in 10 CFR 50.55a. However, in the case of the COL information item in AP1000 DCD Section 3.8.6.2, the ITAAC referred to by the applicant does not fully capture the examination requirements in AP1000 DCD Section 3.8.4.7 that the previous COL information item referred to. The ITAAC addresses examination for leakage and measurement of elevation at two locations before and after filling of the PCS storage tank. AP1000 DCD, Section 3.8.4.7, however, provides additional requirements for examination of excessive cracks in accordance with ACI-349.3R-96. Therefore, in a follow-up RAI, the applicant was requested to include this additional commitment as part of the subject ITAAC or provide the technical basis for excluding it.

In a letter dated September 9, 2009, the applicant agreed to revise the ITAAC in AP1000 DCD Tier 1, Table 3.3-6, to fully capture the examination requirements in AP1000 DCD Section 3.8.4.7 for the PCS storage tank. In addition, the applicant identified that a revision in AP1000 DCD Tier 2, Section 3.8.4.7, was required for testing to be performed to measure the leakage from the PCS storage tank based by measuring the water flow out of the leak chase collection system.

The staff's review of the applicant's September 9, 2009, response determined that the proposed revisions to ITAAC Table 3.3-6 and AP1000 DCD Section 3.8.4.7 are still not consistent. The commitment in AP1000 DCD Section 3.8.4.7 to inspect the PCS tank for significant cracking in accordance with ACI-349.3R-96 is not included in the ITAAC. In addition, the inspection identified in the ITAAC is applicable to the PCS tank boundary and the shield building tension ring while in the case of AP1000 DCD Section 3.8.4.7, the inspection is applicable to the PCS boundary and the shield building roof above the tension ring. The applicant needed to explain whether the inspection would be performed for all three structural regions (PCS tank boundary, shield building roof, and tension ring) and revise both sections of the AP1000 DCD to be consistent. In a follow-up RAI, the staff requested that the applicant address both items discussed above.

In response to the above requests, the applicant's letter dated June 18, 2010, explained that the references to specific standards, such as ACI-349.3R-96, are not included in Tier 1 because this is an established practice in the preparation of Tier 1 information. Since ITAAC Table 3.3-6 in the AP1000 DCD, Revision 15, did not identify the ACI-349.3R-96 standard, but AP1000 DCD Section 3.8.4.7 did, the staff concludes that it is acceptable now to follow the same approach in the current AP1000 DCD.

To address the inconsistency between the proposed revisions to the ITAAC and the AP1000 DCD on the inspection regions, the applicant explained that the design now has steel plates as the outer surface of the tension ring for the enhanced shield building, and concrete cracking in the tension ring region will not be visible; therefore, Table 3.3-6 in the ITAAC will be revised to clarify that the inspection for visible excessive cracking will be performed for the roof

above the tension ring and the PCS tank boundary. Since the proposed revisions to the ITAAC Table 3.3-6 and AP1000 DCD Section 3.8.4.7 are now consistent, the staff concludes that this part of the response is also acceptable.

Based on the above discussion, this is identified as Confirmatory Item CI-SRP3.8.6-SEB1-02, pending revision of the AP1000 DCD to incorporate the mark-ups proposed in the RAI response.

Shield Building COL Items

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and that the application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff concludes that if the items identified above are resolved, the COL information items will meet the applicable acceptance criteria, and that the proposed changes are properly documented in the updated AP1000 DCD. This is based on the additional evaluation report (TR-09) for the containment design adjacent to large penetrations, the inclusion of two ITAAC for the examination of the PCS water storage tank and the as-built summary report, and the existing requirements in 10 CFR 50.55a for the inservice inspection of the containment.

3.8.7 Conclusions

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and that the applicant's application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 foundations as documented in AP1000 DCD, Revision 17, against the relevant acceptance criteria as listed above and in NUREG-0800, Section 3.8.5.

Pending the resolution of the open items and confirmatory items, the NRC staff concludes that APP-GW-GLR-044, TR-85, "Nuclear Island Basemat and Foundation," Revision 1, is acceptable on the basis the analyses and design were performed in accordance with the ACI-349 Code, applicable RGs, and NUREG-0800, Section 3.8.5.

The staff concludes that if the items identified above are resolved, the design of the AP1000 foundations will continue to meet all applicable acceptance criteria.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each applicant would have to address these issues individually.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

The evaluation is performed for AP1000 DCD, Revision 17. The applicant proposed editorial and minor technical changes and clarifications to the section including adding daily load follow operations to the Level A Service Conditions; redefining reactor coolant pump startup and shutdown cases; and defining loading and unloading operations. In addition, the applicant proposed to add WESTEMS design computer code to AP1000 DCD Table 3.9-15 for application of the fatigue analysis of components.

3.9.1.1 Technical Evaluation

AP1000 DCD Tier 2 Subsection 3.9.1.1.1.4 addresses the unit loading and unloading operations associated with power changes of 5 percent per minute between 15 percent and 100 percent power levels. The number of loading and unloading operations is defined as 2,000 each for the 60-year plant design. RAI-SRP3.9.1-EMB1-01 requested the applicant to provide the technical basis for splitting the 2,000 occurrences from the original 19,800 occurrences for the plant loading and unloading at 5 percent of the full power per minute for the normal plant startup/shutdown, and loading resulting from all service levels B, C, and D transients that result in a reactor trip.

In its September 5, 2008, response to RAI-SRP3.9.1-EMB1-01, Westinghouse indicated that when the design transients for the AP1000 were initially established, it was decided to use the unit loading and unloading transient to cover the load follow and increase the number of these transients to cover a daily load follow. It is noted that this was a conservative approach since the load follow transient is less severe than the unit loading and unloading transient. As such, the daily load follow transient will be appropriately addressed rather than assuming the unit loading and unloading transient for most of the load follow requirement. Westinghouse used 2,000 occurrences of unit loading and unloading each to account for shutdowns and the recovery from service level B, C, and D transients. Westinghouse noted that the 2,000 occurrences will cover the approximately 700 total service level B, C, and D transients and 1 (one) per month for loading and unloading each for 60 years. Westinghouse also noted that this frequency is larger than that at currently operating units and is considered bounding. The staff concurs with Westinghouse on the basis of its operating experience and concludes that use of 2,000 occurrences of unit loading and unloading is conservative and acceptable. RAI-SRP3.9.1-EMB1-01 is therefore closed.

AP1000 DCD Tier 2 added a new Subsection 3.9.1.1.1.19, "Daily Load Follow Operations" to Revision 16 to account for the one load follow operation per day that was included as a portion of the plant loading and unloading events for the design transients. RAI-SRP3.9.1-EMB1-02 requested the applicant to provide the basis of how the 17,800 cycles were determined for the daily load follow operations during the plant design of 60 years which with a 90 percent availability factor could result in 19,800 occurrences, and to discuss the basis that the load follow event could not coincide with the plant loading and unloading transients while they might occur at the same time.

In its September 5, 2008, response to RAI-SRP3.9.1-EMB1-02, the applicant noted that the total of unit loading and unloading transients combined with the daily load follow transient is 19,800 transients for 60 years of plant operation based on one transient per day with 90 percent plant availability factor. With the case of reduced power or in a load following mode, the nuclear power plant typically runs on a weekly cycle not a daily cycle. As such, it is assumed that a unit

unloading and a daily load follow event would not occur on the same day. With 2,000 occurrences (each) for unit loading and unloading transients, the remaining 17,800 occurrences are made up of the daily load follow transients. The staff agrees with the applicant's determination to use 17,800 occurrences for a daily load follow transient considering 2,000 occurrences conservative for unit loading and unloading transient as this case is much more severe than the daily load follow transient. Therefore, RAI-SRP3.9.1-EMB1-02 is closed.

As a result of the onsite technical review on October 20, 2008, the NRC staff found that the fatigue analyses for the design of AP1000 seismic Category I components and supports were performed using a computer program called WESTEMS, which is not discussed in the AP1000 DCD subsection 3.9.1.2,"Computer Code Used in Analyses," nor listed in Table 3.9-15, "Computer Programs for Seismic Category I Components." In its response to the staff's RAI-SRP3.9.1-EMB1-03, Westinghouse indicated that the DCD will be revised to add WESTEMS computer program to Table 3.9-15. It also stated that the WESTEMS computer program was not previously reviewed and approved by the NRC staff. On May 26 to 28, 2009, the staff conducted an audit of WESTEMS at Westinghouse headquarters in Monroeville, Pennsylvania. The audit was not completed because not all the documents requested were available at the time of the audit. The follow-up review was completed at the end of September 2009 in the Westinghouse Twinbrook office in Rockville, Maryland.

During the audit, the NRC staff discussed with the applicant the theoretical background, formulation, validation methods, and benchmarking problems pertaining to WESTEMS. The discussions including, in part, the RAIs the staff presented to the applicant during the exit meeting are described in the following paragraphs.

The staff reviewed the WESTEMS basis documents and identified that the stress peak/valley selection option using the stress evaluated with algebraic summation of three orthogonal moment components requires justification. The staff noted that the algebraic summation of three orthogonal vectors is mathematically incorrect and physically meaningless. The staff requested the applicant to provide technical justification for this option in selecting peak and valley times for the fatigue evaluation. This concern was identified as Open Item OI-SRP3.9.1-EMB1-05.

The WESTEMS program provided an option to eliminate peak/valley points during calculation. The staff noted that the computer output should not be modified after executing the program. The staff requested the applicant to provide the configuration control and limitations of the program for this option. This concern was identified as Open Item OI-SRP3.9.1-EMB1-07.

The staff performed an onsite review to discuss/resolve the above mentioned open items. The staff's onsite review summary report (ADAMS Accession Number ML102880708) identified the WESTEMS deficiency.

By a letter dated September 29, 2010, the applicant requested to remove WESTEMS from the DCD markup that adds WESTEMS to Table 3.9-15 of the DCD. In this letter, the applicant stated that the DCD need not include the WESTEMS program because the analyses in question are identified as COL Information Item 3.9-7 in the DCD and are not within the scope of the design certification amendment. The applicant also stated that Westinghouse will use an appropriate analytical tool for performing the aforementioned analyses and the COL applicant

has responsibility to close out the COL Information Item. The staff agreed that the COL applicant is responsible to close out COL Information Item 3.9-7 and fatigue analysis is part of the piping analysis. However, the staff was concerned that this tool should be provided as part of the methodology in the DCD. The staff acknowledged that the methodology available in the DCD in Revision 15 was complete such that the fatigue analysis could be performed without an additional tool. Also, DCD Tier 2, Section 3.9.2.1, states that the Combined License applicant will implement the NRC benchmark program using AP1000 specific problems if a piping analysis program other than those for design certification (PIPESTRESS, GAPPIPE, WCAN, and ANSYS) is used. This statement is marked as Tier 2*. Staff notes that use of a computer code as an analytical tool, as stated above, would require departure from the DCD based on the closure of the COL Item in Section 3.9.8.6 of the application. The closure is discussed in Section 3.12.1.2 of this SE. On the basis that the applicant would return to the previously certified methodology, which was complete, and that any computer code added in the future would require benchmarking, the staff finds this acceptable. Therefore, Open Items OI-SRP3.9.1-EMB1-05 and OI-SRP3.9.1-EMB1-07 are closed.

3.9.1.2 Conclusions

Based on the letter dated September 29, 2010, the NRC staff concludes that the applicant's request to remove WESTEMS from the DCD markup that adds WESTEMS to Table 3.9-15 of the DCD results in no change to the DCD for this item. On the basis mentioned above, the staff determined that all the OIs related to WESTEMS are closed. The staff will evaluate piping design fatigue analysis to ensure piping integrity for safety at the time of COL item closure. The staff concluded that the DCA for Section 3.9.1 is acceptable.

3.9.2 Dynamic Testing and Analysis of Systems, Components and Equipment

3.9.2.3 Preoperational Flow – Induced Vibration Analysis and Testing of Reactor Internals

3.9.2.3.1 Summary of Technical Information

In AP1000 DCD, Revision 17, Section 3.9.2, "Dynamic Testing and Analysis," the applicant proposed changes to reactor internals and analysis. These changes included: addition of a flow skirt to the reactor vessel lower head, addition of neutron panels, relocation of radial support keys and tapered periphery on lower core support plate (LCSP), downcomer excitations and related responses, reduction of core shroud brace thickness, and reactor coolant pump (RCP) induced loads.

3.9.2.3.2 Evaluation

Subsection 3.9.2 of the final safety evaluation report (FSER) describes the AP1000 reactor vessel internals conformance with RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," November 2006, and SRP 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components." The first AP1000 reactor internals design is classified as a prototype, as defined in RG 1.20. However, as stated in WCAP-16716 "AP1000 Reactor Internals Design Changes," the applicant does not consider the AP1000 reactor vessel internals a first-of-a-kind or unique design. Several units

that have operating experience collectively have similar reactor vessel internal design features and are referenced in support of the AP1000 reactor vessel internals design.

The original reference plant for Westinghouse three-loop plant reactor internals flow-induced vibration is H. B. Robinson. The results of vibrations testing at H. B. Robinson are reported in "Westinghouse PWR Internals Vibrations Summary Three-Loop Internals Assurance," WCAP-7765-AR, November 1973. With the addition of neutron panels to the reactor vessel internals design, the applicable referenced plant test has changed from Paluel 1 (no reactor shielding) to Trojan 1 (similar to current neutron panel AP1000 configuration). Westinghouse believes, as stated in WCAP-16716, that the change in referenced plant tests will not impact the conclusions in "AP1000 Reactor Internals Flow-Induced Vibration Assessment Program," WCAP-15949-P, Revision 2, April 2007.

The vibration testing for 17x17 fuel internals and inverted hat upper internals is reported in "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," WCAP-8766, May 1976 and "UHI Plant Internals Vibrations Measurement Program and Pre- and Post-Hot Functional Examinations," WCAP-8516-P, March 1975. The vibration testing of three-loop XL type lower core support structure in DOEL 4 is reported in "Doel 4 Reactor Internals Flow-Induced Vibration Measurement Program," WCAP-10846, March 1985. The vibration evaluations of upper and lower internals assemblies for a four-loop XL plant are reported in "South Texas Plant (TGX) Reactor Internals Flow-Induced Vibration Assessment," WCAP-10865, February 1985. The vibration testing of the core shroud lower internals design is reported in "A Comprehensive Vibration Assessment Program for Yonggwang 4 Nuclear Generating Station, Final Evaluation of Pre-Core Hot Functional Measurement and Inspection Programs," CE Report 10487-ME-TE-240-03, August 22, 1995.

The results of the Doel 3 and Doel 4 reactor internals vibration test programs have been utilized to perform the vibration assessment of the AP1000 reactor internals. The measured responses from Doel 3 and Doel 4 have been adjusted to the higher AP1000 flow rate to support the determination of the expected upper internals and lower internals vibration levels, respectively. The velocity through the core is approximately the same as that of Doel 4.

The results of the Trojan 1 tests showed that the lower internals vibrations are lower with neutron panels than with a circular thermal shield as reported in WCAP-8766.

The NRC staff reviewed the relevant documents as stated above and evaluated the impact of changes in the reactor internals on the vibration evaluations of upper and lower internals assemblies. In addition, the staff reviewed the basis of the Westinghouse contention in WCAP-16716 that there is no impact on the conclusions in the DCD.

3.9.2.3.2.1 Addition of Flow Skirt to the Reactor Vessel Lower Head

The results of the Computational Fluid Dynamics (CFD) calculations using the existing structures in the lower plenum along with the LCSP flow hole geometry indicated that the core inlet flow distribution needed to be adjusted to create a more uniform core inlet flow distribution. The core inlet flow distribution was improved by the addition of a flow skirt to the lower plenum of the reactor vessel.

CFD analyses of numerous configurations of the hardware in the lower reactor vessel have been made with the objective of obtaining a core inlet flow distribution that meets specifications established by the Westinghouse fuel group. It has been determined that flow distributions that meet the requirements are obtained with a flow skirt. A flow skirt is a perforated cylinder in the lower reactor vessel head that is attached to the reactor vessel bottom head. The flow skirt is attached to the lower head of the reactor vessel at the plant site after measurements for machining of the core barrel clevises have been completed. The attachment consists of welds across eight tabs that rest on support lugs provided on the reactor vessel lower head.

There is a circumferential weld between the spherical bottom vessel head and the conical transition to the cylindrical portion of the reactor vessel. The weld is just above the top surface of the flow skirt support lugs. There is some radial clearance between the outside of the flow skirt and the inside surface of the reactor vessel at the circumferential weld location. Examination Category B-N-2 of Section XI, Subsection IWB-2500, provides requirements for the visual (VT-3) examination of "interior attachments beyond the beltline region" of the reactor vessel. Vertical access for a pole-mounted camera is possible around the full circumference of the flow skirt with partial blockage at the four lower radial support keys located on the cardinal axes. It has been judged that the flow skirt and attachment welds could be inspected using VT-3 examinations. If any relevant condition is detected, IWB-3122 (prior to service) or IWB-3142 (inservice) provides options for correcting the condition. The staff reviewed the impact of the welds in generating additional vorticity and turbulence in the lower plenum region.
Based on its review the staff determined that additional information is needed for the staff to complete its review. Several welded joints have been introduced as a result of the addition of the flow skirt, as stated earlier. In RAI-SRP3.9.2-EMB1-07, the staff requested the applicant to discuss the potential for generation of vortices in the region of the flow skirt due to the presence of these welded joints as well as the flow skirt itself and the potential adverse effects on the response of other internals components. The applicant was also requested to discuss any tests related to the evaluation of the flow skirt performance.

In its June 20, 2008, response (ADAMS Accession Number ML081760193), Westinghouse stated, "Any vortices in this region would be proportional in size to the minimum open dimension between the vessel and the flow skirt. This will be on the order of 0.376 in. Any vortices generated will therefore be too small and of too high a frequency (frequency is proportional to velocity divided by vortex dimension) to be of concern. If anything, the flow skirt will tend to dissipate any larger vortices that may be produced by the flow around the radial keys. The fact that the flow skirt makes the lower plenum flow field more uniform is an additional benefit. Because of this, there is a diminished possibility of large velocity gradients entering the lower plenum from the vessel down comer. Lower velocity gradients (greater flow uniformity) also diminish the probability of large vortex-formation. Flow skirts of similar design have been successfully used in operating System-80 plants. A scale model flow test, which includes the flow skirt and its connections to the reactor vessel, is planned as a confirmatory test."

Based on its review, the staff finds that the applicant has provided a reasonable and satisfactory explanation for a dimished likelihood of large vortex formation in the lower plenum region and Open Item OI-SRP3.9.2-EMB1-07 is closed.

3.9.2.3.2.2 Addition of Neutron Panels

To provide flexibility in the core design over the life of the plant, end-of-life reactor vessel fluence calculations were made assuming a radial core power distribution of higher power fuel assemblies in the outmost peripheral locations than in a normal low leakage core. To maintain the end-of-life reactor vessel fluence values at less than the maximum allowed in RG 1.99, neutron panels were attached to the outside diameter of the core barrel. The resulting reactor vessel fluence is 8.9E19 n/cm2 (E > 1.0 MeV) at the end of the 60-year life. Neutron panels have been used on the recent Westinghouse reactor internals designs. They reduce the reactor vessel fluence at the circumferential locations that have the highest fluence values and provide a relatively rigid structure that has a smaller downcomer cross-sectional area than a full cylinder.

The neutron panels are located at four circumferential locations where fuel assemblies are closest to the reactor vessel (0, 90, 180, and 270 degrees). Each pad covers ~30 degrees circumferentially and extends over the entire length of the active core region (14 ft). The pads are contoured to minimize the impact on the downcomer annulus flow area and to reduce the probability of vortex generation in the downcomer.

Based on its review the staff determined that additional information was needed for the staff to complete its review. In RAI-SRP3.9.2-EMB1-02 the staff requested Westinghouse to discuss the potential fluid forces created by the redesigned neutron panels and their potential effects on the flow-induced vibration (FIV) excitation of the core barrel/core shroud. In its June 20, 2008, response, Westinghouse stated "The circumferential extent of the neutron panels was limited to correspond to the high vessel fluence levels, and thus minimize the flow blockage in the downcomer. The neutron panels are tapered circumferentially (following the reduction in fluence level) to minimize the flow area reduction. In addition, the reactor vessel inside diameter was increased by two inches over the core elevations when the panels were added. This results in a net flow area increase of 4 percent relative to the vessel-core barrel downcomer flow area before the panels were added. The lower average downcomer velocity is expected to offset the effects of the turbulence added by the neutron panels."

Based on its review, the staff finds that the applicant has provided a satisfactory explanation of how the additional effects of turbulence due to the neutron panels are neutralized. Therefore, the concerns related to RAI-SRP3.9.2-02 are resolved and the addition of the neutron panels is likely to have no detrimental effects.

3.9.2.3.2.3 Relocation of Radial Support Keys and Tapered Peripheral on the LCSP

The four lower radial support keys for the core barrel are currently located 45 degrees from the cardinal axes. There is also a spherical radius on the outer diameter of the LCSP. Core inlet flow distribution and reactor vessel pressure drop results from computational fluid dynamics (CFD) computer analysis showed that the core inlet flow distribution and the reactor vessel pressure drop were acceptable with a 6-degree slope on the outer diameter of the LCSP. Having the slope instead of the spherical radius on the outer diameter of the LCSP results in sufficient room for the radial support keys to be relocated to the cardinal axes, which is the preferred location. This relocation of the radial support keys eliminates the potential for interference with the core shroud attachment studs and nuts at the 45-, 135-, 225-, and 315-degree locations.

Based on its review the staff finds that relocation of the radial support keys and providing a tapered surface instead of a spherical one has no detrimental effects and is therefore acceptable.

3.9.2.3.2.4 Downcomer Excitations and Related Responses

The nozzle region of the reactor vessel has not been changed so that the entering flow turbulence excitations do not change. The addition of the neutron panels and the increase in the inside (and outside) diameter of the reactor vessel over the core elevations, since the original calculations have been made, change the overall area of the downcomer slightly. The reactor vessel inside diameter below the nozzle has been increased. The flow area including the addition of the neutron panels, increased vessel diameter, and different specimen basket design is increased by approximately 4 percent. This tends to offset the turbulence and increase in local velocities generated by the presence of the neutron panels. Due to the addition of a flow skirt to the lower head of the reactor vessel, the excitations of the structures in the lower vessel head plenum are likely to be lower which also contribute to a lower core barrel vibration level.

Based on its review the staff determined that additional information was needed for the staff to complete its review. Therefore, in RAI-SRP3.9.2-EMB1-10, the staff requested Westinghouse to provide analytical or test data to quantitatively validate this statement that the increase in the increase flow area by 4 percent is expected to offset the turbulence and increase in the local velocities generated by the presence of the neutron panels.

In its June 20, 2008, response, Westinghouse stated that all previous test data show that, for a given geometry and inlet flow pattern, the turbulence excitation decreases-usually by an exponent greater than 2-with decreased flow rate. The staff finds this response satisfactory and acceptable because Westinghouse has provided quantitative data to satisfy staff's concern. Therefore, concerns related to RAI-SRP3.9.2-EMB1-10 are considered resolved.

Based on its review the staff finds that the changes in the vessel diameter, addition of the flow skirt and the presence of the neutron panels will have no detrimental effects on the downcomer excitations and related responses. These changes are, therefore, acceptable.

3.9.2.3.2.5 Reduction of Core Shroud Brace Thickness

Design modifications have been evaluated for the AP1000 core shroud subsequent to the analyses discussed above. The modification is to thin the core shroud braces to reduce thermal stresses. The staff concluded that this modification will not have a detrimental effect on the structural integrity of the core shroud and is therefore acceptable.

3.9.2.3.2.6 Reactor Coolant Pump-Induced Loads

RCP-induced forces are included in the responses reported in Section 7.7.2 of WCAP-15949-P Revision 2. A calculation to predict the pressure differences across the various reactor vessel internals components due to RCP pulsations was performed. However, since the original acoustic calculation using the ACSTIC code was completed, several design changes were made to the AP1000 reactor vessel and reactor vessel internals as discussed above. Specifically, the reactor vessel diameter was increased, the lower core restraints were

relocated, neutron panels were added, specimen baskets were redesigned and relocated, and a flow skirt was added. To evaluate the impact on predicted pressure differences due to the previously noted design changes, an updated ACSTIC calculation was completed.

The updated calculation performs a similar analysis at hot full-power as the original calculation while considering the previously noted design changes. Additionally, the updated calculation also considers the hot functional test (HFT) conditions, including the absence of the core with 25 percent of the core pressure drop simulated near the exit of the LCSP. Consistent with the original calculations, three frequency ranges were evaluated with all RCPs in-phase and with two RCPs out of phase with the other two. The three frequency ranges are ±10 percent of the rotating speed frequency, the first blade passing frequency and the second blade passing frequency. The impact of the results of the updated calculation have been addressed in the individual component analyses for the guide tube, upper support column, core barrel, and core shroud.

The reactor internals were evaluated for the RCP startup conditions shown in Table 5-9a of WCAP-15949-P. The updated reactor conditions are shown in Table 5-9b of WCAP-15949-P. The updated conditions are less severe since the time to reach hot standby is the same for the new and old conditions but the flow rates during heat-up are lower for the new conditions. Therefore, fluid velocities are lower for the updated startup conditions than for the evaluated startup conditions. Lower flow rates would result in lower flow turbulence loads. Since the calculated high-cycle fatigue factors of safety are greater than one, the staff concluded that the AP1000 internals are adequately designed.

Based on its review as discussed above, the staff determined that it needed additional information to complete its review. Therefore, the staff requested Westinghouse to provide this information in the areas of concerns.

In RAI-SRP3.9.2-EMB1-01, the staff requested the applicant to describe the design and modeling of the core barrel/upper core plate as they relate to FIV structural dynamic analysis. The staff also requested the applicant to discuss the uncertainty associated with the modeling of the support interface employed in the modal analysis of the support. In its June 20, 2008, response, Westinghouse stated that the upper core plate is modeled as a part of the upper internals in the system model. The gaps between the upper core plate (and core shroud) slots and the alignment plates mounted on the core barrel are also modeled. To ensure that the entire range of possible gaps between the upper core plate and the core barrel alignment plates is evaluated, time-history analyses were performed with various sets of gaps (upper core plate, top core shroud plate, and core barrel lower supports). Table 6-9 in WCAP-15949-P, Revision 2 (Reference 1), shows the gaps modeled and the resulting loads. The resulting highest load was used in the structural analysis.

The staff finds the applicant's response reasonable and acceptable. Also, AP1000 DCD Section 3.9.2.3 was revised. Therefore, concerns related to RAI-SRP3.9.2-EMB1-01 are resolved.

3.9.2.3.2.7 Evaluation of WCAP-15949-P Revision 2

The staff's review and acceptance of WCAP-15949-P Revision 1 is documented in Subsection 3.9.2.3 of the FSER for the AP1000 (NUREG-1793). The additional information in WCAP-15949-P Revision 2 includes information to justify that there will be no impact on the vibration evaluation of the reactor internals as a result of the changes in the standard design. The Staff's review in this safety evaluation includes this additional information. A preoperational HFT is to be carried out on the first AP1000 reactor internals, classified as a prototype, per requirements of NRC RG 1.20, Revision 2. The AP1000 reactor internal design is the latest product of evolutionary changes to three-loop plants, starting with H. B. Robinson as the first prototype and the most recent ones being Doel 3 and Doel 4 (3XL), as described in Section 1.2 of WCAP-15949-P Revision 2. The significant design changes in the AP1000 reactor internals relative to the Doel 3 and Doel 4 designs are described in Section 3 of WCAP-15949-P Revision 2. The plant and scale model tests associated with each prototype (including the upper internal test of Doel 3 and the lower internal test of Doel 4) are summarized in Section 4, which also demonstrates the consistency among the various Westinghouse plant and scale model tests. The sources of the flow-induced vibration, considered in Section 5, of WCAP-15949-P Revision 2 are the following:

- Flow turbulence
- Reactor coolant pump (RCP) related
- Turbulence excitation of system fundamental acoustic mode
- Vortex shedding

In Section 5 of this WCAP, forcing functions simulating the various excitations are developed through correlation with the 3XL and other plant and scale model test data and put on AP1000 system models and sub-models. The results, in terms of peak stresses, on the various AP1000 critical components are presented in Section 6 and summarized in Table 2-1. Westinghouse has developed detailed CFD and finite-element models of both the 3XL and the AP1000 reactor vessel and internals designs as discussed in Sections 5 and 6 of this report. The 3XL finite-element model is used to calculate vibratory-induced deflections, and the calculated values are compared to applicable plant test data taken during the Doel 4 HFT. The finite-element modeling techniques are refined to accurately predict the Doel 4 test results, and these modeling techniques are applied in the AP1000 model. The CFD model was used to determine the steady-state flow loads on the upper internals components. Section 7 presents the detailed plan for the preoperational HFT and Section 8 presents the pre- and post-hot functional inspection program.

There is no instrumentation between the upper end of the core shroud and the lower core support plate. In RAI-SRP3.9.2-EMB1-03 the staff requested the applicant to discuss the rationale for and the location of instrumentation to provide predicted stresses and also provide the value and location of the maximum stresses for the core barrel/core shroud assembly. In its June 20, 2008, response Westinghouse stated, "A detailed description of the internals model is provided in WCAP-15949, Revision 2. The instrumentation is designed to provide adequate information to describe the vibration time histories and modal content. In the case of the core barrel, the beam modes can be inferred from the core barrel flange strain gages. The fundamental shell modes of the core barrel cover the entire length, the approximate mid point being at the top of the core shroud where three radially sensitive accelerometers are mounted."

The staff finds the rationale for the panel location of the instrumentation reasonable and acceptable. With regard to the locations of the maximum stresses and adequacy of the instrumentation, Westinghouse stated..."the motions are defined by an assembly model. Where needed, sub-models are made to accurately define local, maximum stresses. Detailed core shroud models and sub-models are used to define maximum vibratory stress levels in the core shroud. Similarly, for the core barrel, models are used to define stresses at key locations such as core barrel flange (dominantly beam mode-induced stresses), and shell mode stresses) and barrel shell lower core support plate stresses (includes vertical motion-induced stresses). The strain gages and other transducers are located such that they are not in an extremely high gradient area and so that, with the analytical models they can adequately define the vibration so that maximum stresses can be determined from the analytical models. The maximum stresses for the core barrel/core shroud are provided in Table 2-1 of WCAP-15949. The maximum core barrel stress is at the core barrel wall to core barrel flange interface. The maximum core shroud stress is at the corner of the panel."

Based on its review of the above response, the staff finds that the instrumentation supported by the structural model (which is supported by the calculated versus measured mode shapes and natural frequencies) is adequate to define the maximum stresses due to flow and RCP-induced vibration. Therefore, the concerns related to RAI-SRP3.9.2-03 are resolved.

In WCAP-15949, Table 5.3, "Comparison of calculated and measured 3XL responses," it is stated that the accelerations are considered to be influenced by accelerometer pressure sensitivity and that vertical vibration content in the core barrel strain gages is difficult to ascertain because of masking by other contributors. Therefore, in RAI-SRP3.9.2-EMB1-04, the staff requested Westinghouse to discuss (a) how the vibration content affects the strain gage data, (b) how associated conversion factors from 3XL to AP1000 are affected, and (c) the uncertainties in the conversion factors.

In its response, Westinghouse stated, "The strain gages are used to measure mean and oscillatory reactor internal responses. For example, in the core barrel flange strain gages, the oscillatory content includes contributions from core barrel beam modes, the vertical modes of the core barrel, and the shell modes of the core barrel. Supported by the core barrel analytical model and data from other transducers, the contribution of the various modes can be determined. This information is used to support the determination of the maximum stress in the core barrel flange.

During the 3XL hot functional vibration testing, it was observed that the accelerometer data included an unexpected magnitude of response at a particular frequency that was postulated to be due to system pressure pulsations. The accelerometer pressure sensitivity was confirmed by the accelerometer vendor. It is considered that this was adequately recognized in the interpretation of the 3XL data. The 3XL test data are used only to benchmark the analytical methods used to predict AP1000 responses, primarily the CFD based prediction of core barrel vibration. There are no conversion factors used in developing the AP1000 responses, since all of the AP1000 predictions are from analytical models."

Based on its review of the above response, the staff finds that Westinghouse has provided a satisfactory response to the staff's concerns related to how the vibration content affects the strain gage data, associated conversion factors from 3XL to AP1000 are affected, and the

uncertainties in the conversion factors. Therefore, the concerns related to RAI-SRP3.9.2-EMB1-04 are resolved.

The overall methodology for estimating the vibration forces and using these forces to predict the response of the reactor internals is outlined in Figure 5-1 of WCAP-15949. In RAI-SRP3.9.2-EMB1-05, the staff requested Westinghouse to describe the methodology for determining bias errors and uncertainties associated with data obtained from various sources for evaluating AP 1000 reactor internals responses.

In its response, Westinghouse stated, "The transducers are calibrated prior to use. From this calibration, the voltage conversions at the temperature that the data were acquired are applied. Any uncertainty in the factors that convert voltages to physical units will also be recognized. It is also noted that expected and measured responses were similar in past tests. In view of these factors, it is considered that bias errors and uncertainties are less than the minimum margin to allowable values-presently 0.2 for AP1000 (per WCAP-15949-P Revision 2, Table 2-1)."

The staff finds the applicant's explanation for justifying the bias errors and uncertainties as being less than 0.2 to be reasonable and satisfactory. Therefore, concerns related to RAI-SRP3.9.2-EMB1-05 are resolved.

The FSER related to the certification of the AP1000 standard design (NUREG-1793), discusses the evaluation of WCAP-15949-P Revision 1 in Section 3.9.2.3 of the FSER. In RAI-SRP3.9.2-EMB1-06, the staff requested Westinghouse to discuss and summarize the significant additional information/items provided in WCAP-15949-P Revision 2, dated June 2007.

In its response Westinghouse stated that the most significant changes between Revision 1 and Revision 2 of WCAP-15949 are the addition of the neutron panels, the reactor vessel diameter increase in the core region, the revised specimen basket arrangement, and the addition of a flow skirt to the reactor vessel. The overall conclusion that the vibration amplitudes are sufficiently low for structural adequacy of the AP1000 reactor internals has not changed. Westinghouse also provided an itemized list of changes between WCAP-15949-P, Revision 1 and Revision 2, in the RAI response. The staff reviewed this itemized list of changes and concerns related to RAI-SRP3.9.2-EMB1-06 are resolved.

Past experience related to testing of reactor internals indicates that instrument failures do occur during testing. Thus, it is prudent to provide redundancy in the data acquisition process. Therefore, in RAI-SRP3.9.2-EMB1-08, the staff requested the applicant to discuss the redundancy in the instrumentation proposed for the AP1000 reactor internals preoperational test program.

In its response Westinghouse stated, "Some redundancy is included in the number, location, and types of transducers installed during the Hot Functional Test program. For example both accelerometers and strain gages are installed on the core barrel, which provides some redundancy in the event that an individual transducer would fail. In previous prototype tests conducted by Westinghouse, the instrument failures were not of sufficient quantity to preclude drawing the needed conclusions.

The transducers are installed on the reactor internals and subjected to known static and dynamic inputs prior to the Hot Functional Test. These calibration tests relate displacements to measured strains and accelerations and this data is used to interpret the mean flow loads and flow-induced vibration amplitudes. The operability of these transducers is also verified during these static and dynamic calibration tests. In addition, some redundancy is included in the interpretation of the results in that a narrow band response centered on a particular frequency can be associated with a particular mode and the damping of that mode. This enables the stress distribution associated with this mode to be used to completely describe the stresses related to this mode.

Based on its review of the applicant's response as discussed above, the staff finds that there is adequate redundancy in the instrumentation and satisfactory calibration procedures are in place. Therefore, the concerns related to RAI-SRP3.9.2-EMB1-08 are resolved.

In RAI-SRP3.9.2-EMB1-09, the staff requested the applicant to provide the following topical reports, which relate to preoperational test programs for the Trojan 1 and Doel 4 plants that are referenced in the AP1000 DCD Revision 17: (1) WCAP-8766, Verification of Neutron Pads and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant, (2) WCAP-10846, Doel 4 Reactor Internals Flow-induced Vibration Measurement Program. Additionally, the applicant was requested to provide test data from the core shroud at the Yonggwang 4 plant, which is relevant to the evaluation of the AP1000 reactor internals.

In its June 20, 2008, response, Westinghouse provided the two WCAP reports and the Yonggwang core shroud test report for staff review at the Westinghouse Rockville, MD office. The staff reviewed these documents. The results of the Doel 3 and Doel 4 reactor internals vibration test programs were used to perform the vibration assessment of the AP1000 reactor internals. The measured responses from Doel 3 and Doel 4 were adjusted to the higher AP1000 flow rate to support the determination of the expected upper internals and lower internals vibration levels respectively. The velocity through the core is approximately the same as that of Doel 4. Based on its review the staff was satisfied that the applicant had used an acceptable methodology to perform the vibration assessment of the AP1000 reactor internals. The results of the Trojan 1 tests confirmed that the lower internals vibrations are lower with neutron panels than with a circular thermal shield as reported in WCAP-8766.

The staff is satisfied with the results, and concerns related to RAI-SRP3.9.2-EMB1-09 are resolved.

An acoustic analysis of the primary coolant loop has been provided in Section 5.1.3.1 of WCAP-15949. The impact of the results of the updated calculations has been addressed in the individual component analyses for the guide tube, upper support column, core barrel, and core shroud. The reactor internals were evaluated for the RCP startup conditions shown in Table 5-9a. The updated reactor conditions are shown in Table 5-9b of WCAP 15949. It is noted that the updated conditions are less severe since the time to reach hot standby is the same for the new and old conditions but the flow rates during heat-up are lower for the new conditions. Therefore, fluid velocities are lower for the updated startup conditions than for the evaluated startup conditions. Lower flow rates would result in lower flow turbulence loads. Westinghouse therefore concludes that there would be no overall impact due to the design changes.

In order to evaluate the impact on predicted pressure differences due to the design changes, an updated acoustic analysis using the computer code ACSTIC, was performed. However, simplifying assumptions were made in the acoustic modeling. The staff contended that the conclusions are not necessarily valid unless adequate justification is provided that the uncertainties associated with the ACSTIC calculation have been taken into consideration. The staff requested Westinghouse in RAI-SRP3.9.2-EMB1-11 to discuss how the uncertainties associated with acoustic analysis were factored into the results of the updated calculations.

In its response, Westinghouse stated, "The uncertainties associated with the ACSTIC calculation were considered by employing a general design basis in which the RCP-related responses are taken to be coincident with natural frequency if the natural frequency is within ±10 percent of the RCP excitation frequency. The calculated maximum forces from this resonance condition were then utilized in the reactor internals component structural evaluation."

The staff finds the applicant's response reasonable and acceptable, and concerns related to RAI-SRP3.9.2-EMB1-11 are resolved.

Based in its review of WCAP-15949-P Revision 2, and Revision 17 of the AP1000 Design Control Document, Section 3.9.2.3, the staff finds that there is no overall impact due to the design changes.

3.9.2.3.3 Conclusion

This report supplements the FSER for the AP1000 standard plant design. The FSER was issued by the NRC as NUREG-1793 in September 2004 to document the NRC staff's technical review of the AP1000 design. With the closure of OI-SRP3.9.2-EMB1-07 documented in this SER, the staff concludes that the applicant has provided sufficient information to satisfy 10 CFR Parts 50 Appendix A, GDC 1 and 4 with regard to the dynamic testing and analysis of systems, structures, and components.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

3.9.2.4.1 Introduction

In Revision 16 to the AP1000 DCD, Westinghouse proposed to address COL Information Item 3.9-2 pertaining to irradiation-assisted stress-corrosion cracking (IASCC) and void swelling susceptibility evaluations for reactor internal core support structure materials.

In Section 3.9.2.4 of the FSER for the AP1000 standard plant design (NUREG-1793), the NRC staff identified COL Action Item 3.9.2.4-1, in which the COL applicant will provide the design reports for the reactor internal core support structures including a final stress analysis conforming to the design provisions of the ASME Code, Section III, Subsection NG. The following section addresses the adequacy of the analyses for the reactor internals (RIs) for IASCC and void swelling phenomena.

AP1000 Standard COL TR-12, APP-GW-GLR-035, Revision 0, was provided by Westinghouse under Westinghouse Report WCAP-16620-P, Revision 0, "Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted

Stress Corrosion Cracking (IASCC) and Void Swelling for the AP1000 Plant," (hereafter designated as TR-12) dated July 31, 2006. TR-12 addresses AP1000 COL Information Item 3.9-2 pertaining to IASCC and void swelling in reactor internal core support structure materials for the AP1000 plant. COL Information Item 3.9-2 corresponds to AP1000 Design Control Document (DCD), Tier 2, Subsection 3.9.8.2 (DCD Subsection 3.9.8.2), Revision 15 and Action Item 3.9.2.4-1 from the NRC FSER on the AP1000 plant. COL Information Item 3.9-2 is addressed in a proposed revision to DCD Subsections 3.9.8.2 and 3.9.9. The NRC staff reviewed the information provided in TR-12, including the proposed changes to DCD Subsections 3.9.8.2 and 3.9.9. The revised DCD subsections are included in Revision 16 to the AP1000 DCD. The staff's findings regarding TR-12 are summarized below.

In TR-12, Westinghouse addressed the provisions of COL Information Item 3.9-2 pertaining to IASCC and void swelling susceptibility evaluations for reactor internal core support structure materials for the AP1000 plant. Westinghouse proposed to revise COL Information Item 3.9-2, in part, through the implementation of Revision 16 to DCD Subsection 3.9.8.2. In Revision 15 to the AP1000 DCD, Subsection 3.9.8.2, the COL Information Item stated:

> Combined License applicants referencing the AP1000 design will have available for NRC audit the design specifications and design reports prepared for ASME Section III components. COL applicants will address consistency of the core support materials relative to known issues of irradiation-assisted stress corrosion cracking and void swelling. [*The design report for the ASME Class 1, 2, and 3 piping will include the reconciliation of the as-built piping as outlined in subsection 3.9.3. This reconciliation includes verification of the thermal cycling and stratification loadings considered in the stress analysis discussed in subsection 3.9.3.1.2.*]

It should be noted that TR-12 only addresses the second sentence of DCD, Revision 15, Subsection 3.9.8.2. The other sentences in this revision to DCD Subsection 3.9.8.2 are addressed in separate AP1000 Standard COL TRs.

In Revision 16 to the AP1000 DCD, Westinghouse proposed to address the COL Information Item on a generic basis and revise Subsection 3.9.8.2 as it relates to IASCC and void swelling to state:

> The consistency of the reactor internal core support materials relative to known issues of irradiation-assisted stress corrosion cracking and void swelling has been evaluated and addressed in APP-GW-GLR-035 (Reference 21).

Revision 16 to DCD Subsection 3.9.8.2 specifically references TR-12 (i.e., APP-GW-GLR-035) as the technical basis for the evaluation of IASCC and void swelling phenomena in AP1000 reactor internal components. In addition to the above, Revision 16 to the AP1000 DCD adds the following reference (Reference No. 21) for TR-12 to DCD Subsection 3.9.9, "References":

> 21 APP-GW-GLR-035, "Consistency of Reactor Vessel Internal Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking and Void Swelling for the AP1000 Plant," July 2006.

3.9.2.4.2 Background

IASCC is an age-related degradation mechanism where materials exposed to high levels of neutron radiation become more susceptible to SCC with increasing neutron fluence. The current consensus is that susceptibility to IASCC is a significant concern for austenitic stainless steel and nickel-based alloy reactor internal components in both Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs). This is due to the fact that these components are exposed to elevated neutron fluence levels over the lifetime of the plant. The exact mechanisms for IASCC damage in reactor internal components are not well known. However, numerous studies suggest that IASCC results from the synergistic effects of irradiation damage to the material, changes in the local coolant-water chemistry, and the stress state in the component.

Irradiation-induced void swelling is an environmental degradation phenomenon that can affect reactor internal structural alloys exposed to high levels of neutron radiation. Void swelling is characterized by an increase in a component's volume due to the formation of voids as a result of neutron irradiation at elevated temperatures. Void formation occurs due to the migration and condensation of lattice vacancies in response to radiation-induced displacement of atoms from their lattice sites. Void swelling becomes more pronounced at higher structural temperatures due to higher diffusion rates. Some amount of swelling can occur in virtually all structural alloys under sufficiently high conditions of neutron fluence and temperature. However, austenitic stainless steels and nickel-based alloys, the primary alloys used in reactor internal core support components, are known to be susceptible to void swelling earlier and faster due to the multiple slip systems and close-packed nature of their face-centered cubic crystal structure. As many PWRs age, void swelling behavior in austenitic stainless steel and nickel-based alloy reactor internal components has become the subject of increasing attention. Excessive void swelling can lead to dimensional instability of the component and significant decreases in fracture toughness. It could also influence or contribute to the susceptibility of the component to IASCC, stress relaxation, and irradiation embrittlement.

3.9.2.4.3 EPRI Topical Report MRP-175

The U.S. Nuclear Power Industry is conducting ongoing studies of IASCC and void swelling phenomena in reactor internal structural components. The IASCC and void swelling data that have been accumulated thus far were summarized in a report issued by the Electric Power Research Institute (EPRI), Topical Report MRP-175, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)," dated December 2005. This report provided screening criteria and their technical bases for the age-related degradation evaluation of PWR reactor internal component items.

Appendix B of MRP-175 addressed IASCC in PWR reactor internal components and the establishment of an IASCC threshold and screening criteria for determining susceptibility to IASCC behavior. The report provided a comprehensive review of the open literature and industry operating experience regarding IASCC in American Iron and Steel Institute (AISI) Type 304 and 316 austenitic SSs; the differences in IASCC behavior of cold-worked versus solution-annealed SSs; and IASCC behavior in nickel-based alloys. In general, this review confirmed that IASCC may be a significant concern for reactor internal components during later stages in

plant operating life. Although the exact mechanisms for IASCC are not yet known, the MRP-175 review cited numerous studies conclusively demonstrating that both the stress state in reactor internal components and radiation damage caused by increasing neutron fluence levels during plant service will result in increased susceptibility to IASCC. The review pointed to various studies indicating that radiation hardening is directly linked to IASCC. Radiation-induced segregation, a phenomenon of accelerated solute diffusion brought about by radiation-induced increases in vacancy concentration, was also cited as a possible contributor to IASCC. The IASCC studies and limited industry operating experience reviewed by MRP-175 were used as a basis for recommending IASCC screening criteria based on stress levels in the component and accumulated radiation-induced displacement damage, quantified in units of displacements per atom (dpa). For a given material exposed to specific radiation energy spectra, increasing neutron fluence values correlate directly with increasing dpa levels.

The MRP-175 review cited studies suggesting that thermo-mechanical history and chemical composition can potentially have a significant impact on IASCC resistance in austenitic stainless steel materials. In particular, cold-working has been shown to be potentially favorable for delaying the onset of radiation damage at lower damage levels (less than 10 dpa). This phenomenon has been attributed to the presence of a high density of dislocations for trapping radiation-induced point defects, thereby delaying the development of the microstructure responsible for radiation hardening. However, at higher damage levels (greater than 10 to 20 dpa), studies indicate that both solution-annealed and cold-worked materials attain the same degree of radiation hardening. Studies also indicate that differences in bulk alloy composition among various austenitic stainless steel reactor internal components can potentially have varying effects on IASCC initiation and progression. The higher nickel content of Type 316 was cited as a contributor to its greater resistance to radiation damage, compared with Type 304 SS.

Oversize solutes such as titanium and niobium may also contribute to IASCC resistance by serving as trapping sites for point defects. Overall, MRP-175 concluded that, while IASCC susceptibility among various austenitic stainless steel materials is recognized to be affected by thermo-mechanical history and chemical composition, no consistent or quantitative correlation has yet been established. Thus, it was determined that a conservative set of IASCC screening criteria should be applied to all stainless steel alloys.

Section B.3 of MRP-175 stated that, based on numerous studies of IASCC phenomena, certain neutron fluence levels are a necessary precondition for the occurrence of IASCC in reactor internal components. For austenitic SSs, the MRP-175 review of data in the literature points to a conservative fluence threshold for IASCC in PWR reactor internal components of approximately 7×10^{20} n/cm^2 (E > 1.0 MeV), or a radiation damage level of about 1 dpa. However, the only known PWR IASCC incidents, observed in European PWR baffle bolts, have indicated an IASCC threshold level of approximately 2×10^{21} n/cm^2 (E > 1.0 MeV), or about 3 dpa. Additional evidence for the higher IASCC damage threshold was provided by studies which determined that IASCC initiation at 1 dpa can only occur under extremely high strain conditions (40 percent decrease in laboratory specimen cross section); such high strains are not representative of conditions in PWR reactor internal components. Further studies demonstrated that an IASCC damage threshold of 3 dpa existed for various heats of cold-worked 316 SS, where stress levels in lab specimens exceeded the yield strength for the material. Based on these studies and the incidents that were observed in European PWR baffle bolts, the MRP-175 report concluded that 3 dpa represented a reasonable consensus estimate of the IASCC

damage threshold for austenitic stainless steel reactor internal components. However, the MRP-175 report emphasized that, at the current time, the understanding of IASCC is not sufficiently advanced to suggest a definitive IASCC fluence or radiation damage threshold that is universally applicable to all PWR reactor internal materials.

Despite significant uncertainty regarding a precise IASCC threshold and the definitive prediction of IASCC susceptibility in PWR reactor internal components, the studies reviewed in the MRP-175 report point to a definite correlation of IASCC behavior with neutron fluence and stress levels in the component. Figure B-1 of MRP-175 presented curves, based on IASCC laboratory studies, depicting the stress level required for specimen failure by IASCC as a function of radiation damage, in dpa. A recommended IASCC screening curve was presented in Figure B-3 of MRP-175. This screening curve was derived by shifting the empirical curve for long term IASCC failure downward (to more conservative stress levels) to account for the observed baffle bolt failures in Europe. MRP-175 recommended that this lower bound IASCC screening curve be utilized at this time for developing IASCC screening criteria for PWR reactor internal components where radiation damage levels exceed 3 dpa.

Appendix G of MRP-175 addressed void swelling in PWR reactor internal components and recommended void swelling screening criteria. In general, MRP-175 found that void swelling may be a significant concern for reactor internal components in PWRs because it produces volume and dimensional changes that could potentially result in distortions within structural components as well as changes in fracture toughness properties. The MRP study of void swelling phenomena found that when volume changes in the material exceed approximately 5 percent, significant increases in embrittlement associated with the void swelling start to occur.

Furthermore, the MRP review of fast reactor data found that when volume changes in the material due to void swelling exceed 10 percent, the tearing modulus for 300-series stainless steels is dramatically reduced and falls to zero at room temperature, corresponding to severe embrittlement with little energy required for crack propagation.

Based on a comprehensive review of the literature and industry operating experience regarding void swelling behavior in austenitic SSs, MRP-175 concluded that void swelling behavior in reactor internal components is primarily influenced by structural temperature in the component and accumulated radiation damage (dpa level), with components becoming more susceptible to void swelling at higher temperature and damage levels. Studies also demonstrate that neutron flux (corresponding to the dpa rate) can affect void swelling behavior, with lower dpa rates resulting in greater swelling for a given accumulated dpa level. However, the effect of dpa rate on void swelling in PWRs has not been well quantified, and MRP-175 cited several other void swelling studies that did not observe a strong effect.

Numerous studies cited by MRP-175 have reported that other factors are known to affect void swelling behavior in reactor internal components. Void swelling (VS) data demonstrate that cold work has the beneficial effect of prolonging the void swelling incubation period, due to the elevated concentration of dislocations acting as traps for point defects in cold-worked materials. Chemical composition of stainless steel alloys is also known to affect void swelling behavior. For instance, nickel and chromium content strongly affect vacancy diffusivity, and therefore, the onset of VS. On this basis alone, Type 304 stainless steel always swells more than Type 316 with the same thermo-mechanical starting state. Stress is generally regarded as a factor that

accelerates swelling, although it is not thought to be an important factor for most PWR applications. MRP-175 also pointed to various studies showing that a high helium content or helium production rate can affect void swelling behavior. Several studies suggest that the presence of preexisting helium gas bubbles may prolong the incubation period of void swelling under high dpa rates in fast reactors. This is thought to be due to helium gas bubbles acting as sinks for point defects, thereby delaying the onset of rapid swelling. However, under normal neutron irradiation conditions in PWRs, various studies have given conflicting results regarding the overall impact of helium on void swelling behavior in reactor internal components. For instance, helium atoms generated as a result of the transmutation of boron during irradiation can increase the swelling rate, as helium atoms combine with vacancy clusters, thereby facilitating void nucleation and growth. Furthermore, the production of helium gas bubbles in components during transmutation could have the net effect of increasing the overall swelling, thereby negating any beneficial effects of vacancy elimination.

MRP-175 suggested that screening of austenitic stainless steel reactor internal components for void swelling should be determined primarily by the structural temperature of the material, the accumulated dpa level, and the dpa rate that the material will experience during service. MRP-175 emphasized that the screening criteria should focus on the volume changes that occur as a result of void swelling behavior because embrittlement and distortion of the component, the primary structural consequence of significant VS, occurs as a result of these volume changes. MRP-175 cited numerous studies suggesting that the onset of VS-induced embrittlement occurs at a local void swelling percentage of approximately 5 percent. It was therefore recommended that void swelling of one-half this level (~2.5%) should necessitate further examination of the component. If it can be ascertained that local swelling in a component would never approach 2.5 percent, then void swelling is not a concern.

To date there have been no reports of PWR reactor internal components showing significant distortion or failures as a result of VS. The only PWR void swelling data comes from baffle bolts removed for IASCC evaluations. Very minor void concentrations were observed with transmission electron microscopy (TEM) in several baffle bolts removed from Point Beach, Unit 1; Farley, Unit 1; and Tihange (Belgium), Unit 1. MRP-175 summarized the results of these evaluations. The highest localized void fraction was estimated at 0.24 percent in one of the bolts removed from the Tihange plant. All other local void swelling measurements were significantly less, with half of the measurements showing no voids present. Furthermore, 0.24 percent void swelling would not be expected to significantly impact structural performance. Based on these data, MRP-175 determined that for austenitic stainless steel reactor internal components, localized regions with structural temperatures less than 320 °C (608 °F) and projected damage levels less than 20 dpa (~ 1.3×10^{22} n/cm^2, E > 1.0 MeV) would be expected to experience local void swelling levels of less than 2.5 percent. This was recommended as the preliminary criterion by which void swelling in the component may be ruled out. MRP-175 stated that localized regions in reactor internal components with structural temperatures greater than 320 °C (608 °F) and projected damage levels greater than 20 dpa (~ 1.3×10^{22} n/cm^2, E > 1.0 MeV) should be analyzed to determine the percentage increase in void fraction using the best currently available predictive equation developed by industry studies of void swelling behavior for 304 series stainless steel – Equation G-2 from MRP-175. This equation correlates the percentage increase in void concentration with temperature, dpa level, and dpa rate. If this equation yields a predicted void swelling percentage greater than 2.5 percent, then further functionality evaluations for the component are necessary.

3.9.2.4.4 Evaluation

The evaluation of AP1000 reactor internal components for potential susceptibility to IASCC and void swelling was addressed in TR-12. Section 1.2 of TR-12 provided a brief discussion of known issues of IASCC and void swelling in the currently-operating PWR fleet. Westinghouse indicated that reactor internal components in currently-operating Westinghouse plants have not exhibited significant IASCC or void swelling issues to date based on inservice inspections (ISIs) performed in accordance with the requirements of the ASME Code, Section XI. However, other PWR vendors have reported limited IASCC in reactor internal bolting applications for several PWR plants in Europe. Results from detailed inspections of cold-worked Type 316 stainless steel baffle bolts from Farley, Unit 1 (a Westinghouse three-loop design) showed no signs of cracking after 17 effective full power years (EFPY) of facility operation. The estimated neutron fluence exposure for these baffle bolts is 20 dpa.

Based on the IASCC studies and data that have been accumulated thus far, the known parameters directly affecting the onset and progression of IASCC in reactor internal structural components are peak stress level in the component and cumulative exposure to neutron radiation (neutron fluence) during plant service. For VS, the known parameters affecting its onset and progression are peak structural temperature in the component and neutron fluence. Therefore, screening of reactor internal components for potential susceptibility to IASCC and void swelling requires that these parameters be determined. Section 2 of TR-12 briefly discussed the calculation of these parameters for use in IASCC and void swelling screening evaluations. Westinghouse determined that IASCC screening would be based upon the peak stress to which a reactor internal component is subjected at full hot power. The peak stresses were said to be comprised of the "membrane stress intensity with additions due to bending and stress concentrations, steady state thermal stress additions, and high-cycle fatigue components." Westinghouse stated that transients do not need to be considered for the IASCC stress calculations. The peak stress levels for each of the reactor internal components were provided in Table 2-1 of TR-12. The projected end-of-life (EOL) radiation damage levels for each of the reactor internal components were provided in Table 2-2. These damage levels were expressed in units of dpa. Table 2-3 listed the estimated structural temperatures for each of the reactor internal components during normal operation.

Section 3 of TR-12 discussed the screening of reactor internal core support structure components for potential susceptibility to IASCC. The components were evaluated through the use of a set of PWR-specific screening criteria based on stress state in the component and damage level. These screening criteria are essentially a set of threshold levels of damage level and stress, such that if the specific EOL damage level and structural stress levels for a given component are found to be below the screening criteria threshold levels, it could be concluded that IASCC would not be an applicable degradation mechanism for the component during the design life of the plant. Conversely, if the EOL damage level and structural stress levels for a component are found to be greater than or equal to the screening criteria threshold levels, IASCC is considered to be a potential degradation mechanism during the service life of the component. According to TR-12, satisfaction of the IASCC screening criteria (i.e., exceeding the stress and damage level threshold values) does not imply that IASCC will absolutely occur; rather it should be considered as a potential degradation mechanism.

The IASCC screening criteria used in TR-12 are as follows:

- For EOL damage level < 3 dpa, IASCC is not considered applicable for any stress conditions.
- For EOL damage level ≥ 3 dpa, IASCC may be applicable for specific ranges of damage level and stress. These ranges are defined as follows:
- For 3 dpa ≤ EOL damage level ≤ 10 dpa, IASCC is considered applicable if stress ≥ 427.5 MPa (62 ksi).
- For 10 dpa < EOL damage level ≤ 20 dpa, IASCC is considered applicable if stress ≥ 317.2 MPa (46 ksi).
- For 20 dpa < EOL damage level ≤ 40 dpa, IASCC is considered applicable if stress ≥ 206.8 MPa (30 ksi).
- For the three dpa ranges above, it is implied that if the component does not meet the applicable stress threshold, IASCC would not be considered applicable.

Table 3-1 of TR-12 evaluated the peak stress and EOL damage level for each of the reactor internal core support structure components against the above IASCC screening criteria to determine whether or not any of the components would be susceptible to IASCC. Although a number of components have a projected EOL damage level greater than 3 dpa, none of these components have peak stresses that exceed the IASCC threshold levels for stress listed above. It was therefore concluded that IASCC is not a potential degradation concern for the reactor internal core support structure components for the design life of the AP1000 plant.

Section 4 of TR-12 discussed the screening of reactor internal core support structure components for potential susceptibility to radiation-induced void swelling. The potential susceptibility of components was evaluated through the use of a PWR-specific screening criterion based on the structural temperature in the component during normal operation and EOL damage level. The void swelling screening criterion used in Section 4 of TR-12 is as follows:

> If the structural temperature for a component is greater than or equal to 320 °C (608 °F) during normal reactor operation, and the EOL damage level equals or exceeds 20 dpa, then void swelling has a potential to occur.

Section 4 of TR-12 invoked the criterion above to screen all reactor internal core support structure components for susceptibility to void swelling. Although several of the reactor internal core support structure components are listed as having either a structural temperature or an EOL damage level that is greater than the applicable threshold, none of the components were listed as having both structural temperature and EOL damage level greater than or equal to the above thresholds. Accordingly, the results of this screening led Westinghouse to the conclusion that none of the reactor internal core support structure components for the AP1000 plant are susceptible to void swelling for the design life of the plant.

Based on its initial review of the above information regarding the screening of AP1000 reactor internal components for potential susceptibility to IASCC and VS, the staff determined that additional information was required to complete its evaluation. In an RAI issued on January 18, 2007, the staff requested supplemental information concerning the IASCC and void swelling screening methodology. RAI questions 1, 3, 4, 5, 6, 8, 10, 11, 12, 13, and 14 addressed the IASCC screening methodology. RAI questions 2, 7, 9, and 15 addressed the void swelling screening methodology. Westinghouse provided responses to these RAI questions by letter dated May 2, 2007 (ADAMS Accession Number ML071270244).

In RAI Question 1, part a (RAI 1a), the staff requested that Westinghouse clarify whether the IASCC and void swelling screening criteria were meant to be specific for the AP1000 reactor design or were meant to be applied to PWR environments, regardless of PWR design. In its response to RAI 1a, Westinghouse stated that the IASCC and void swelling screening criteria are generic for all PWR environments and may be applied to reactor internal components regardless of design. The staff found that this response adequately resolved RAI 1a because Westinghouse clarified the applicability of the IASCC and void swelling screening criteria.

In RAI 1b, the staff requested Westinghouse to confirm whether the IASCC screening criteria from Section 3 of TR-12 were established using the lower bound IASCC screening curve developed by EPRI in Figure B-3 of the MRP-175 report. In its response to RAI 1b, Westinghouse confirmed that the IASCC screening criteria in TR-12 were established using the lower bound IASCC screening curve developed by EPRI in Figure B-3 of the MRP-175 report. The staff found that this response adequately resolved RAI 1b because Westinghouse provided the requested statement regarding the bases for the IASCC screening criteria in Section 3 of TR-12.

In RAI 1c, the staff requested that, if the IASCC screening criteria in Section 3 of TR-12 were established based on the lower bound IASCC screening curve from Figure B-3 of the MRP-175 report, Westinghouse provide justification, based on environmental and material similarity, regarding how these IASCC screening criteria are applicable to reactor internal components for the AP1000. In its response to RAI 1c, Westinghouse stated that the materials specified for the AP1000 reactor internal components are similar to those used in the currently-operating Westinghouse three-loop extended length design. Operating parameters are also similar. IASCC screening of AP1000 reactor internal components was based on the same criteria (the lower bound IASCC screening curve from Figure B-3 of MRP-175) as those used for IASCC evaluations of reactor internal components in these operating reactors. Furthermore, the MRP-175 IASCC screening curve was developed as a generic lower bound curve for austenitic stainless steel reactor internal components in PWR environments, and its application was not intended for any specific set of material conditions (e.g., amount of cold-work, solution annealing, trace element composition). With respect to environmental similarity, the MRP-175 screening curve is based on radiation damage and stress level for the component, and according to the current understanding of IASCC, these are the two known environmental parameters directly affecting the onset and progression of IASCC behavior. Therefore, the IASCC screening curve in Figure B-3 of the MRP-175 report is applicable to the AP1000 reactor internal components, based on environmental and material similarity. Accordingly, the staff found that RAI 1c is resolved.

In RAI 1d, the staff requested that Westinghouse indicate whether reactor internal components that do not meet or exceed the IASCC screening criteria in TR-12 (i.e., components that do not meet or exceed the threshold stress and damage levels for IASCC) would ever be considered susceptible to IASCC. In its response to RAI 1d, Westinghouse stated that ongoing license renewal and life extension activities at operating Westinghouse reactors will develop new data concerning aging effects and aging management in reactor internal components. It is possible that new data may necessitate the consideration of IASCC in reactor internal components currently not considered susceptible to IASCC. However, at the present time, the IASCC screening criteria in Section 3 of TR-12 are applied for the purpose of determining whether or not a given AP1000 reactor internal component is susceptible to IASCC behavior during the operating life of the plant. Since none of the AP1000 reactor internal components have peak stress and EOL damage levels that meet or exceed the IASCC threshold levels from Section 3 of TR-12, none of the components are currently considered susceptible to IASCC. The staff found that this response adequately resolved RAI 1d because Westinghouse clearly stated how it applied the screening criteria for determining susceptibility to IASCC.

In RAI 2, the staff requested Westinghouse to confirm whether the void swelling screening criterion from Section 4 of TR-12 was established based on the void swelling screening recommendation developed by EPRI in Section G.7 of the MRP-175 report. The staff further requested in RAI 2 that Westinghouse provide justification, based on environmental and material similarity, regarding how the void swelling screening criterion is applicable to reactor internal components for the AP1000. In its response to RAI 2, Westinghouse confirmed that the void swelling screening criterion from Section 4 of TR-12 is based on the void swelling screening recommendation of MRP-175. With respect to the applicability of the MRP-175 void swelling screening recommendation to AP1000 reactor internal components, Westinghouse stated that the materials specified for the AP1000 reactor internal components are similar to those used in the currently-operating Westinghouse three-loop extended length design. Operating parameters are also similar. Screening of AP1000 reactor internal components for void swelling was based on the same criterion (the void swelling screening recommendation from Section G.7 of MRP-175) as that used for void swelling evaluations of reactor internal components in these operating reactors. Furthermore, the MRP-175 void swelling screening recommendation was intended to be generic for austenitic stainless steel reactor internal components in PWR environments, and its application was not intended for any specific set of material conditions (e.g., amount of cold work, solution annealing, trace element composition). With respect to environmental similarity, the MRP-175 void swelling screening recommendation is based on neutron fluence and peak structural temperature for the component, and based on the current understanding of VS, these are the two known environmental parameters directly effecting the onset and progression of void swelling behavior. Therefore, the void swelling screening recommendation from Section G.7 of the MRP-175 report is applicable to the AP1000 reactor internal components, based on environmental and material similarity. Accordingly, the staff found that RAI 2 is resolved.

In RAI 3, the staff requested further detail regarding how the peak stresses for the various reactor internal components in Table 2-1 of TR-12 were determined. The staff also requested, in RAI 3, that Westinghouse elaborate on why stresses arising from thermal transients were not considered in the peak stress calculations. In its response to RAI 3, Westinghouse stated that these stresses represented peak stress levels for normal operation. Finite element techniques were used in the computation of these stresses, and stress concentration factors were applied

as appropriate. The reported stresses were intended to be conservative for IASCC screening of reactor internal components. With respect to consideration of thermal transients, Westinghouse indicated that the screening criteria stress levels (based on the MRP-175 IASCC screening curve) were developed for comparison with normal operating peak stress levels, and normal operating peak stress levels do not include stresses due to transient conditions. However, these stress levels do account for steady-state thermal stresses arising from temperature gradients within the reactor internal components during normal operation. Westinghouse emphasized that temperature gradients in reactor internal components are a steady-state phenomenon caused by the surrounding reactor coolant system temperatures and internal heat generation within reactor internal components due to gamma heating; these factors are known to result in steady-state temperature gradients and thermal stresses within reactor internal components during normal operating conditions. The staff found that this response adequately resolved RAI 3 because Westinghouse adequately clarified its methods for computing the peak stresses for the reactor internal components. Furthermore, Westinghouse conclusively defined these stresses as peak operating stresses that do not account for transient conditions and provided adequate justification for why transients were not considered in their computation. Therefore, the staff found that RAI 3 is resolved.

In RAI 4, the staff requested Westinghouse to define end-of-life (EOL) for the projected radiation damage levels in Table 2-2 of TR-12 in terms of the total effective full power years (EFPY) of facility operation. In its response to RAI 4, Westinghouse stated that EOL for the AP1000 design is considered to be 55.8 EFPY of facility operation. Therefore, the damage levels in Table 2-2 of TR-12 are projected out to 55.8 EFPY of facility operation. The staff found that this response adequately resolved RAI 4.

In RAI 5, the staff requested Westinghouse to discuss how ISI will be conducted for the reactor internal components during the operating life of the AP1000 plant. In its response to RAI 5, Westinghouse stated that ISI of reactor internal components during plant operating life will be driven by applicable codes and standards, as required by NRC regulations. At present, a VT-3 visual examination of all accessible surfaces of reactor internal core support structure components is required by the ASME Code, Section XI. These examinations must be conducted once during each 10-year ISI interval. Such visual examinations are currently performed using remotely controlled submersibles, underwater crawlers and/or pole-mounted cameras. The staff found that this response adequately resolved RAI 5 because Westinghouse adequately specified how ISI will be conducted for reactor internal components during the operating life of the AP1000 plant.

In RAIs 6 and 7, the staff requested Westinghouse to discuss how the EOL damage levels and estimated structural temperatures from Tables 2-2 and 2-3 of TR-12 were determined for the reactor internal components. In its response to RAI 6, Westinghouse stated that a radiation model of the reactor vessel and internal components was created and two distinct axial power distributions were utilized to determine damage levels in dpa. The higher damage level from the two core power distributions was listed for each reactor internal component in Table 2-2. In its response to RAI 7, Westinghouse stated that detailed finite element thermal calculations were performed to determine the structural temperatures reported in Table 2-3. These calculations accounted for the effects of gamma heating using two core power distributions. The distribution resulting in the highest component temperature was utilized and temperatures at localized regions within the components were evaluated. The highest localized temperature for the

component during normal reactor operation was listed in Table 2-3. As with the peak operating stresses listed in Table 2-1, the structural temperatures listed in Table 2-3 represent peak temperatures during normal operation because the void swelling temperature threshold in Section 4 of TR-12 (based on the screening recommendation of MRP-175) was developed for comparison with normal operating temperature levels in reactor internal components. The staff found that these responses adequately resolved RAIs 6 and 7 because Westinghouse adequately clarified its methods for computing the EOL damage levels and structural temperatures from Tables 2-2 and 2-3 of TR-12. Furthermore, the staff found that these stated methods were appropriate for calculating temperature and damage levels for use in screening reactor internal components for IASCC and VS.

In RAI 8, the staff requested Westinghouse to discuss whether there are any localized areas within any reactor internal component that could be exposed to damage levels that exceed the IASCC screening criteria from Section 3.1 of TR-12. In its response to RAI 8, Westinghouse stated that the EOL damage level calculations accounted for localized areas in the reactor internal components. As such, the damage levels reported in Table 2-2 of TR-12 represent that maximum projected damage level based on the highest localized exposure in each component. Therefore, the staff found that RAI 8 is resolved.

In RAI 9, the staff requested Westinghouse to further explain how it screened certain reactor internal components for susceptibility to VS. Specifically, the staff noted that Section 4 of TR-12 concludes that void swelling is not a significant degradation mechanism for any of the reactor internal components in the AP1000 plant. This conclusion was apparently based on the fact that none of the reactor internal components met the void swelling screening criterion, as invoked in Section 4 of TR-12, which stated that if the structural temperature for a component is greater than or equal to 320 °C (608 °F) during normal reactor operation, and the EOL damage level equals or exceeds 20 dpa, then void swelling has a potential to occur. The staff reviewed the damage level projections and structural temperature levels listed in Tables 2-2 and 2-3 and noted that, while none of the components are listed as having both damage level *and* temperature greater or equal than the above temperature and damage level threshold values, several components are listed as having either temperature *or* damage level greater than the applicable threshold. Therefore, the staff requested Westinghouse to explain how it was determined that void swelling was not an applicable degradation mechanism for these components.

In its response to RAI 9, Westinghouse stated that the TR-12 void swelling screening criterion was based on the recommendations in the MRP-175 report and, as such, it requires that both temperature *and* damage level be greater than or equal to the above threshold levels. The staff did not agree with this interpretation of the void swelling screening recommendation from the MRP-175 report and, therefore, found that this response did not adequately resolve RAI 9. By letter dated July 11, 2007, the staff issued a second RAI on this subject in order to address screening of reactor internal components for void swelling where either temperature or damage level meet or exceed the above threshold levels. In this RAI, the staff indicated that the recommended void swelling screening criterion from the MRP-175 report was misinterpreted by TR-12 when applied to reactor internal components that met or exceeded only one of the two thresholds (temperature or damage level). The staff stated the position that void swelling may be a potential concern for reactor internal components if either temperature or damage level exceeds its applicable threshold. This position is justified because of the hypothetical situation

where one of these parameters is significantly greater than the threshold, and the other is only marginally less. For such a situation, it would be unacceptable to dismiss the possibility of void swelling in the component only because just one the two thresholds had been exceeded. Therefore, the staff requested that Westinghouse justify why the several components that were listed in TR-12 as having either temperature or damage level greater than the applicable threshold were not deemed susceptible to VS.

In its second response to RAI 9, dated August 21, 2007, Westinghouse provided an analysis for demonstrating that there are no significant void swelling concerns for the components listed in TR-12 as having either temperature *or* damage level greater than the applicable threshold level. Westinghouse demonstrated that none of the components in question meet the hypothetical situation proposed by the staff, where one of the parameters (temperature or damage level) is significantly greater than the threshold, and the other is only marginally less. For the components with structural temperatures exceeding the 320 °C (608 °F) void swelling threshold, all of the EOL damage levels for these components are far below the 20 dpa damage threshold for VS, and the calculated structural temperatures are only slightly greater than the 320 °C (608 °F) threshold. One component, the core barrel inner wall, has a projected EOL damage level that is slightly greater than the 20 dpa threshold; however the calculated structural temperature is significantly less than the 320 °C (608 °F) threshold. Westinghouse further demonstrated that these components are extremely unlikely to experience any significant void swelling during the operating life of the plant by applying equation G-2 from MRP-175 for calculating the predicted void swelling percentage. Application of this void swelling equation to the dpa and temperature values listed in Table 2-2 and 2-3 of TR-12 and the dpa rate based on 55.8 EFPY of facility operation yields void swelling percentages of less than 0.10 percent for all of these components. MRP-175 recommended further examinations of reactor internal components for void swelling behavior are necessary only if the predicted void swelling percentage, based on this equation, approaches 2.5 percent. Therefore, Westinghouse adequately demonstrated that void swelling is not a significant concern for any of these reactor internal components (or any other AP1000 reactor internal component) based on the current void swelling data and predictive models. Accordingly, the staff found that RAI 9 is resolved.

In RAI 10, the staff requested Westinghouse to reconcile differences between the 3 dpa damage threshold for IASCC susceptibility established in TR-12 and IASCC neutron fluence thresholds established in other reports. Specifically, the staff noted the IASCC neutron fluence threshold from a previous Westinghouse report, WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals," is 1×10^{21} n/cm^2 (E > 0.1 MeV). Additionally, the Babock and Wilcox report, BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," stipulates a neutron fluence threshold of $1\text{-}2 \times 10^{21}$ n/cm^2 (E > 0.1 MeV). The staff noted in RAI 10 that, according to MRP-175, the 3 dpa threshold from TR-12 is roughly equivalent to an accumulated neutron fluence value of 2×10^{21} n/cm^2 (E > 1.0 MeV) for austenitic stainless steel materials. In its response to RAI 10, Westinghouse stated that 3 dpa was the recommended IASCC damage threshold from the MRP-175 report, and the MRP-175 screening criteria represent a consensus opinion of the EPRI MRP expert panel. The recommended 3 dpa damage threshold superseded the previous two Westinghouse reports. The staff found this response adequately resolved RAI 10 because MRP-175 determined that 3 dpa represents a reasonably conservative consensus value for an IASCC damage threshold for reactor internal components in PWR environments.

In RAI 11, the staff requested Westinghouse to discuss whether the 3 dpa damage threshold for IASCC in TR-12 was determined taking into consideration the effect of thermo-mechanical history (e.g., prior cold work, annealing, etc.) in reactor internal components that are exposed to neutron fluence levels less than 6.7×10^{21} n/cm^2 (E > 1.0 MeV). This question related to a statement from Section B.1.1 of MRP-175 referencing studies indicating that thermo-mechanical history may affect the onset of IASCC in reactor internal components exposed to these neutron fluence levels. In its response to RAI 11, Westinghouse stated the IASCC screening criteria, as applied to the reactor internal components in TR-12, are based on the screening recommendations of MRP-175. The MRP-175 IASCC screening recommendations are generic for all austenitic stainless steel materials in PWR environments. As such, the IASCC screening criteria and 3 dpa damage threshold were not developed based on any specific state of cold work (or any other prior thermo-mechanical preconditioning) in the material. While the amount of prior cold work had been shown to potentially delay the onset of IASCC at fluence levels less than 6.7×10^{21} n/cm^2 (E > 1.0 MeV), Westinghouse stated the AP1000 reactor internal components were screened using the MRP-175 screening recommendations without regard to the components' thermo-mechanical history. Westinghouse further stated it is not anticipated that a material's degree of cold work will necessitate screening criteria be different from the criteria recommended by MRP-175. The staff found this response adequately resolved RAI 11 because Westinghouse adequately explained why the MRP-175 IASCC screening recommendations were applied irrespective of the thermo-mechanical history of the components. The staff's justification for acceptance of MRP-175 recommendations for generic screening of AP1000 reactor internal components for IASCC (irrespective of the components' thermo-mechanical history) is provided below.

In RAI 12, the staff requested Westinghouse to discuss whether the 3 dpa damage threshold for IASCC in TR-12 was determined taking into consideration the effect of differing chemical composition for the various reactor internal components. This question related to a statement from Section B.1 of MRP-175 referencing studies indicating differences in bulk alloy composition of elements such as silicon, nickel, niobium, titanium, and boron, among various austenitic stainless steel reactor internal components can have varying effects on IASCC initiation and progression. In its response to RAI 12, Westinghouse stated the IASCC screening criteria, based on the generic screening recommendations of MRP-175, did not consider variations in the elemental composition among the various reactor internal components. As with the case above concerning the potential effect of components' thermo-mechanical history, the staff found this response adequately resolved RAI 12 because Westinghouse adequately explained why the MRP-175 IASCC screening recommendations were applied irrespective of the components' specific elemental composition. The staff's justification for acceptance of MRP-175 recommendations for generic screening of AP1000 reactor internal components for IASCC (irrespective of the components' specific elemental composition) is provided below.

In RAI 13, the staff requested that Westinghouse discuss whether the 3 dpa damage threshold for IASCC in TR-12 is applicable to reactor internal components fabricated from nickel-based alloys, such as alloy X-750 and alloy 690. In its response to RAI 13, Westinghouse indicated IASCC studies reviewed in MRP-175 have shown that the IASCC resistance of nickel-based alloy X-750 is approximately the same as for Type 304 and 316 austenitic SSs. Furthermore, AP1000 reactor internal reactor internal components fabricated using nickel-based alloys will be exposed to a projected EOL damage level of, at most, 0.04 dpa. Since this damage level is far below the 3 dpa IASCC damage threshold, IASCC is not considered to be a relevant

degradation mechanism for these components. The staff found this response adequately resolved RAI 13 because Westinghouse adequately addressed IASCC screening of reactor internal components fabricated from nickel-based alloys.

Studies have shown crevice corrosion may be enhanced in reactor internal components due to the production of oxidizing ions in component crevices during exposure of reactor coolant to neutron radiation. Therefore, in RAI 14, the staff requested that Westinghouse discuss whether the effects of crevice corrosion were taken into consideration in screening AP1000 components for IASCC. In its response to RAI 14, Westinghouse stated the IASCC screening criteria do not explicitly address the effects of crevice corrosion in reactor internal components. However, crevice corrosion is prevented or controlled in AP1000 reactor internal components through the use of hydrogen overpressure, which minimizes the adverse effects of any oxygen that may be present due to heat-up or cool-down of the reactor system. Furthermore, crevice locations in AP1000 reactor internal components have been designed to allow flushing to prevent stagnation, a key contributor to crevice corrosion. The staff found this response adequately resolved RAI 14 because Westinghouse addressed how crevice corrosion would be mitigated in AP1000 reactor internal components.

Transmutation products such as helium are known to play an important role in VS. In order to reduce overall interfacial energy, helium atoms will combine with vacancy clusters, thereby facilitating void nucleation and growth. Section G.1 of MRP-175 states a potentially important aspect of void swelling in PWRs arises from transmutation of trace amounts of boron, preexisting in most austenitic SSs, to produce lithium and helium. Section G.1 of MRP-175 indicates at low neutron exposure (~10^{21} n/cm^2 thermal), almost all Boron-10 (20 percent of natural boron preexisting in trace quantities in most SSs) will be converted to lithium, producing helium in the process. Since the original concentration of boron in austenitic stainless steel reactor internal components is not generally reported in certified material test reports, it is difficult to assess the concentration of helium in the reactor internal components. In RAI 15, the staff requested Westinghouse to address whether the void swelling screening criterion from Section 4 of TR-12 accounts for the effects of helium on void swelling in stainless steel reactor internal components. In its response to RAI 15, Westinghouse stated the void swelling screening criterion, based on the generic screening recommendations of MRP-175, did not explicitly consider the effects of helium. The staff found this response adequately resolved RAI 15 because Westinghouse explained the MRP-175 void swelling screening recommendations were applied irrespective of the components' helium content. The staff explained its acceptance of the void swelling screening evaluation for the AP1000 reactor internal components (irrespective of the components' potential helium content) in the discussion of the applicant's responses to RAI 9 above.

The acceptance of MRP-175 screening recommendations would provide a basis for setting IASCC screening criteria in TR-12. There are currently limited data to support an all-encompassing set of IASCC screening criteria that can be generally applied to reactor internal components in PWRs. Furthermore, MRP-175 has referenced studies showing variability in chemical composition, microstructural characteristics, and thermal-mechanical history between similar alloys may result in differing stress and fluence thresholds for IASCC. MRP-175 cited numerous documents both in the nuclear power industry and the open literature identifying a variety of possible threshold values for IASCC susceptibility and, therefore, a definitive, all-encompassing set of IASCC screening criteria is not likely to exist. In its response to NRC staff

comments regarding these issues, EPRI acknowledged that exact threshold values for IASCC are expected to depend on variables, such as chemical composition, microstructural properties, and thermo-mechanical history. However, EPRI stated the IASCC screening recommendations of MRP-175 represent a consensus based on the limited amount of available data, and the IASCC screening criteria are considered to be conservative for general application to IASCC evaluations of reactor internal components in PWRs. As such, MRP-175 concluded the IASCC screening criteria were appropriate for evaluating stainless steel reactor internal components to determine their susceptibility to IASCC behavior.

The staff found the limited amount of data does support the MRP-175 conclusions regarding the conservatism of the IASCC screening criteria from Section B.3 of MRP-175. Therefore, although it may be impossible to absolutely rule out the possibility of IASCC, because reactor internal components are deemed not susceptible according to the MRP-175 screening criteria, significant IASCC behavior would not be expected for the AP1000 reactor internal components because the peak operating stresses and projected EOL damage levels for these components fall significantly below the MRP-175 screening criteria threshold levels. Furthermore, any age-related degradation of reactor internal components due to IASCC would be gradual, and the ASME Code, Section XI requirements for ISI of reactor internal components will be sufficient for capturing any age-related degradation that may occur due to IASCC phenomena.

Based on the above considerations, the staff determined Westinghouse had adequately addressed the staff's concerns, as documented in the above RAIs, regarding the IASCC and void swelling screening methodologies. Therefore, the staff found Westinghouse had appropriately evaluated the AP1000 reactor internal components for susceptibility to IASCC and void swelling in TR-12. Furthermore, the staff agreed with the conclusions in TR-12 regarding the determination that IASCC and void swelling are not projected to be significant degradation concerns for the reactor internal components in the AP1000 plant.

The staff determined the TR-12 conclusions regarding the evaluation of reactor internal components for IASCC and void swelling meet the requirements of ASME Section III based on the MRP-175 screening criterion as reported in TR-12 and is fully represented in Subsections 3.9.8.2 and 3.9.9 of the AP1000 DCD, Revision 16. Therefore, the staff found the DCD changes, as proposed by Westinghouse in TR-12, are acceptable, and AP1000 COL Information Item 3.9-2 is resolved. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes incorporated into Revision 16 contribute to the increased standardization of the certification information in the AP1000 DCD and, thus, meet 10 CFR 52.63 (a)(1)(vii).

3.9.2.4.5 Conclusions

The staff finds the evaluation of the AP1000 reactor internal components for IASCC and void swelling meets the requirements of 10 CFR 50.55a by meeting the ASME Section III based on the MRP-175 screening criterion as reported in TR-12 and resolution of IASCC aspects of COL Information Item 3.9-2 is acceptable.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Introduction

The staff evaluation was first performed for AP1000 DCD, Revision 16, and TR-134, Revision 5, which were issued as part of AP1000 Design Certification Amendment application. The staff subsequently included AP1000 DCD, Revision 17, in its evaluation when the latest revision was issued by Westinghouse.

3.9.3.2 Technical Evaluation

DCD Tier 2, Subsection 3.9.8.2, addresses the combined license information for the design specifications and reports for the major ASME Code, Section III components. In DCD Tier 2, Revision 17, COL Information Item in Subsection 3.9.8.2, Westinghouse stated the design specifications and design reports for the major ASME Code, Section III components are available for NRC audit via the TRs listed in Table 3.9-19. It is also stated that design specifications and selected design analysis information are also available for ASME Code, Section III valves and auxiliary components. Westinghouse letter dated February 8, 2008, states design specifications and design reports for most major components and auxiliary equipment and valves will be available for NRC review in July 2008. In RAI-3.9.3-EMB2-01, the staff requested Westinghouse to verify the schedule provided in the letter is still valid.

By letter dated June 26, 2008, Westinghouse stated review of design specifications and as-designed design reports for ASME Code Section III components provides a means for the NRC to verify the design commitments in the DCD are being implemented appropriately. This review permits some level of verification during the review of the COL applications. The ultimate check on the proper implementation of design requirements for ASME Code Section III components are the ITAAC require as-built design reports for the ASME Code, Section III components. Westinghouse stated it has a substantial amount of design information available for NRC review. This information is sufficient for the NRC to start its review and support the conclusion the ASME Code, Section III components are in compliance with the commitments in the DCD. The remaining design information needed to complete the NRC verification of the design criteria and methodologies in the DCD will be available in the short term consistent with the schedule for review of the AP1000 Design Certification amendment.

Westinghouse stated since the information needed to address the COL Information Item on design specifications and design reports will be complete during the review of the AP1000 Design Certification amendment there is no need for a Design Acceptance Criteria or Design ITAAC on the design of ASME Code Section III components. Westinghouse revised its approach of resolving the component design issue, and stated the revision of the COL Information Item in DCD Revision 16, Subsection 3.9.8.2, was based on the expectation the design information available at the time was sufficient for the NRC to reach a conclusion as to the implementation of the design requirements. Westinghouse stated the amount of information now available for NRC review is much more developed and robust. The COL information will be revised to reflect that sufficient information will be provided to the NRC to complete its verification of the implementation of the design commitments. Westinghouse stated it is expected this portion of the COL Information Item will be satisfied and no additional information will be required of the COL applicant. The portion of the COL Information Item was restated as

a COL holder item is related to the as-built reconciliation of thermal cycling and stratification loadings on piping.

Westinghouse stated it has completed the design specifications for the major ASME Code, Section III components in the AP1000. The as-designed design reports and supporting analysis for most of the major components are also complete and available for review. Westinghouse stated the analyses include the use of the updated (six soils case) seismic design spectra. The components that have or will have as-designed design reports ready for review on a schedule to support the NRC preparation of the SER with OIs include: reactor vessel, control rod drive mechanism, steam generator, pressurizer, passive RHR heat exchanger, core makeup tank, accumulator, and reactor internals. Westinghouse stated all of the design specifications for valves and auxiliary equipment are now available for NRC review or will be ready in time to support the preparation of the SER. This includes the motor-operated globe and gate valves. The balance of the design specifications (expected to be one or two) will be available for review well before OIs need to be cleared for the advance SER.

The staff has reviewed the above additional information provided by Westinghouse, and found the commitment and schedule for resolving the COL Information Item to be acceptable. Pending a successful audit for the required design specifications and design reports, the staff would be able to conclude whether the COL Information Item is closed.

In its letter of June 28, 2008, Westinghouse provided a markup of the revised DCD Subsection 3.9.8.2, which now states the following:

> The design specification and design reports for the major ASME Code, Section III components and piping made available for NRC review are identified in APP-GW-GL-002. Design specifications for ASME Code, Section III valves and auxiliary components made available for NRC review are also identified in APP-GW-GL-TBD

In doing so, Westinghouse has proposed to delete Table 3.9-19 from DCD, Tier 2, which lists the TRs summarizing design specification and design reports for ASME Code Section III components and piping. Westinghouse also deleted references to Reference Items 22 through 32 in DCD Subsection 3.9.9, and placed a new Reference 22, APP-GW-GL-002, "Design Specifications and Design Reports for ASME Code, Section III Components and Piping," Westinghouse Electric Company LLC.

The staff has reviewed the above information provided by Westinghouse and concludes that the additional information provided by Westinghouse is acceptable in responding to the staff's request of RAI-3.9.3-EMB2-01. Subsequently, the staff reviewed AP1000 DCD, Revision 17, when it became available, and verified the latest revision of DCD has incorporated all the changes as required.

During October 13 to 17, 2008, the staff conducted an onsite review of the AP1000 component design in relation to the close out of the above COL Information Item in DCD Subsection 3.9.8.2. The purpose of the on-site review was to verify the AP1000 component design was in accordance with the methodology and design criteria described in the DCD, and satisfies the guidance provided in SRP Section 3.9.3 for design specifications and design reports. This

includes verification the design information described in the DCD was adequately translated into documentation for each of the components designed to ASME Code Section III, Class 1, 2, and 3 requirements. A separate staff audit report, dated August 3, 2009, (ADAMS Accession Number ML092150664) documents the detailed on-site review for the design of the AP1000 mechanical components, including valves. During the audit, the staff identified concerns with the reactor vessel J-groove weld design and the additional details on the containment recirculation screen design.

The staff requested in RAI-SRP-3.9.3-EMB2-05 that Westinghouse demonstrate how the Westinghouse methodology meets the ASME Code for the J-groove weld design.

In response to RAI-SRP-3.9.3-EMB2-05, Westinghouse stated it has satisfied the intent of Paragraph NB-3228.5(a) of the ASME Subsection NB. According to Westinghouse, the purpose of the Paragraph NB-3228.5 is to limit potential excessive distortion due to incremental plasticity, sometimes referred to as stress ratcheting. The location where this applied is the J-groove weld between the piping penetration and reactor vessel head. The overstress shown in the design report is caused by the large hoop stress combined with, to a lesser degree, the axial stress. The ratcheting mechanism cannot occur as a result of the hoop stress since it is restrained by the reactor vessel head. The stresses in the radial and axial directions are well within the limits and meet the ASME Code requirements. Therefore, according to Westinghouse, additional plastic analysis in accordance with Paragraph NB-3228.4 is not necessary.

The staff found the Westinghouse response unacceptable, and again asked Westinghouse to provide additional information or detailed information to demonstrate the J-groove weld design meets the ASME Code requirements. The staff's concern was that the design report for the RV head penetrations split the stress components at these locations to justify the satisfaction of the Code requirements. Paragraph NB-3228 is based on stress intensities and does not allow splitting stresses for the purpose of satisfying the Code. Westinghouse did not demonstrate why a plastic analysis is not necessary.

Westinghouse in its response stated that the justification previously provided for meeting the requirements of NB-3228.5 is compatible with ASME Code methodology. According to Westinghouse, the fatigue evaluation for stresses is made for a plane of reference. The fatigue evaluation checks the range of stress intensity values for every potential plane (line) of failure and fatigue usage is determined for a point on that plane using a conservative value of primary plus secondary stress intensity range, for the purpose of determining a conservative value of Ke and therefore a conservative usage factor. As this conservative approach does not satisfy the limits of Paragraph NB-3228.5(a), the Code rules are used to perform a more realistic evaluation using membrane and bending stresses normal to the plane of reference. Using this approach, Paragraph NB-3228.5(a) is met with very large margin. This approach is within the Code rules specified in the Code definitions. This evaluation therefore demonstrates compliance with Paragraph NB-3228.5 and a plastic analysis is not required. The reactor vessel design report and associated stress calculation for the vessel head penetrations will be revised with this discussion.

During April 19 to 21, 2010, the staff conducted an onsite review of Open Item OI-SRP-3.9.3-EMB2-05, Component Supports, and Core Support Structures. Westinghouse, in its plastic

analysis of control rod drive mechanism (CRDM) and vent pipe penetrations, has demonstrated that the design of the vessel head assembly satisfies the ASME Code requirements. On this basis, the staff finds this acceptable and Open Item OI-SRP-3.9.3-EMB2-05 is closed, pending revision to the following documents: (1) WH Report APP-MV01-Z0C-015, "Detailed Analysis of Closure Head and Vessel Flange Region for AP1000 Reactor Pressure Vessel," must be revised to reflect the design change associated with the UMI penetrations, and (2) WH Report APP-MV01-Z0C-019, "Detailed Analysis of Closure Head Penetrations (CRDM, UMI, and Vent Pipe) for AP1000 Reactor Vessel," must be updated to include the results of the plastic analysis for the CRDM and vent pipe penetrations. RAI-SRP3.9.3-EMB2-05 is identified as **Confirmatory Item CI-SRP3.9.3-EMB2-05**.

The staff reviewed the design specification and other supporting documents associated with Containment Recirculation Screens and found several issues are not incompletely addressed in the design specification. In RAI-SRP-3.9.3-EMB2-08 the staff requested Westinghouse to address the following:

(a) According to the design specification, the supplier will provide additional design details, design drawings and requirements. Therefore, the engineering drawings (envelope drawings) of the screen assemblies were not available at the time of site audit or at the Rockville office. Provide these engineering drawings of the screen assemblies for review by the staff.

(b) Provide the following loading conditions and combinations: (i) design and service level A-D loads and load combinations, (ii) fatigue evaluation, and (iii) the origin and the basis of using ±5 psi pressure loading on the IRWST screen from sparger discharge.

(c) Justify the latent debris mass value used for the screen pressure drop component of the structural load on the IRWST and sump screens. Additionally, verify the flow rate through the screen is conservatively calculated.

During April 19 to 21, 2010, the staff conducted an onsite review of Open Item OI-SRP-3.9.3-EMB2-08. As a result of the audit, the staff identified follow-up items that required Westinghouse's response:

1. Screen design reports and detailed design drawings were not available, since they have not yet been provided by the responsible vendor. A set of drawings was reviewed by the staff, but these are categorized as "Envelope Drawings" and not detailed design drawings. However, AP1000 DCD, Tier 1, Table 2.2.3-4, ITAAC 5.a) was added to require review of a report verifying the as-installed screens including seismic load, post accident operating loads, head loss and debris weights.

   Subsequently, the staff confirmed the existing AP1000 DCD, Tier 1, Table 2.2.3-4, ITAAC 5.a) addressed the issue such that this ITAAC allows the staff to inspect the design reports and verify the as-installed screens including seismic load, post accident operating loads, head loss and debris weights. The staff concurred with the ITAAC approach of screen design report. The staff finds that these audit findings satisfied the staff's request for the information in part (a) RAI-SRP-3.9.3-EMB2-08 and follow-up item #1. Therefore, the follow-up item number 1 is closed.

2. The staff questioned the loading on the screen (i.e., how the 0.25 psi pressure drop loading will be added to the 5 psi loading for the screens). Following discussion, Westinghouse agreed the screen head loss component of pressure loading on both the IRWST and Containment screens would be a minimum of 0.25 psi as indicated in a Westinghouse design specification (APP-GW-GLE-002, Impacts to the AP1000 DCD to Address Generic Safety Issue GSI-191, February 2010, p. 39). In addition, Westinghouse stated that it would consider augmenting this minimum head loss to allow for additional margin.

   In response to this follow-up item, documented in Westinghouse's response to OI-SRP3.9.3-EMB2-08, Revision 1, the applicant stated it is very unlikely that sparger actuation will occur coincident with IRWST injection as the sparger actuation comes from the ADS 1, 2, 3 discharge. IRWST injection cannot occur until the pressure in the RCS drops below the hydrostatic pressure exerted on the IRWST water due to the level in the tank. This only occurs after ADS 4 has been actuated. Therefore the 0.25 psi debris differential pressure load is not coincident with 5 psi loading for the screens. These two loads are not included in the same load combination. The staff finds the response satisfies the staff's request for the information in part (b)(iii) RAI-SRP3.9.3-EMB2-08 and the follow-up item number 2. Therefore, the follow-up item number 2 is closed.

3. The staff asked the applicant to confirm the applicability of the 5 psi sparger loading on IRWST screen design. The operation of the IRWST tank spargers leads to a pressure loading on the IRWST screens. An estimate of the sparger pressure loading is used as one component of the loading of both the IRWST and the containment screens. During discussions, the staff asked Westinghouse for a review of the available documents that support the pressure loading of ±5 psia. Westinghouse responded by saying the documentation for this was not available at the time. However, Westinghouse stated the documentation for the magnitude of the sparger pressure loading was available, and that NRC staff had this in their possession.

   In response of this follow-up item, documented in OI-SRP3.9.3-EMB2-08, Revision 1, the applicant stated the 'Sparger Loading' is the maximum value due to actuation of the sparger. The actual shape of the pressure will be sinusoidal shape. The forces in the walls of the IRWST are bounded by a case with a uniform pressure of 5 psi applied to the walls. The actuation of the sparger will occur during discharge of ADS 1, 2, 3 valves. Tests conducted at the ENEA's VAPORE facility showed the maximum pressure exerted on the IRWST walls during a sparger actuation of 400 lbm/s steam. The pure steam blowdown caused the highest pressures exerted on the IRWST floor directly below the sparger arm during sparger actuation. Additionally, the tests simulated a sparger steam flow of 400 lbm/s. This flow more than bounds the actual calculated maximum steam flow of 145 lbm/s for the AP1000 (APP-ME02-Z0C-001 Revision 0). The nominal hydrodynamic load exerted during the above mentioned sparger test was 5 psig. Given the steam flow was more than 2.7 times the actual design flow, this bounds the structural design requirement for DP mentioned above. The staff finds the response satisfies the staff's request for the information in part (b)(iii) of RAI-SRP3.9.3-EMB2-08.

The staff confirmed these support documents are available for NRC review and verified the pressure loading of ±5 psia was acceptable by SRP Section 6.2 evaluation. The staff concurred with Westinghouse's response of the pressure loading of ±5 psia. Therefore, this follow-up item is closed.

4. The staff questioned the potential sloshing of water in the IRWST tank resulting from seismic activity and the magnitude of resulting pressure loading on the IRWST screen structures. Westinghouse did not have a response to whether this potential source of loading had been considered. The follow-up item for Westinghouse is to determine if the sloshing load on the screen should be included.

   In response to this follow-up item, documented in OI-SRP3.9.3-EMB2-08 R1, the applicant stated the sequences for the postulated accidents are such that the seismic event producing the sloshing and the actuation of the sparger are not coincident. Therefore, the sloshing loads need to be included as a load on the screen, but it is not combined with the sparger loadings. The staff finds the response satisfies the staff's request for the information in part (b)(i) of RAI-SRP3.9.3-EMB2-08. However, the applicant should update the load and load combination table of DCD to reflect that the screen needs to be evaluated with the seismic-induced sloshing loads. Therefore, this follow-up item is **Confirmatory Item CI-SRP3.9.3-EMB2-08.**

5. In response to part (c) RAI-SRP3.9.3-EMB2-08, documented in OI-SRP3.9.3-EMB2-08 R1, the applicant stated that the AP1000 debris inventory in containment and the development of supporting containment cleanliness programs have been reviewed by the NRC staff. The screen pressure drop and the flow rate through the screen have been developed in support of responding to GSI-191 issues. This information is documented in APP-GW-GLN-147, Screen Design Report and APP-GW-GLR-79, Verification of Water Sources for Long Term Recirculation Following a LOCA.

The staff confirmed these support documents are available in NRC review, verified AP1000 debris inventory in containment and the development of supporting containment cleanliness programs were acceptable by SRP Section 6.2 SER evaluation. The staff concurred with Westinghouse's response of part (c) RAI-SRP3.9.3-EMB2-08.

The DCD changes proposed in response to RAI-SRP3.9.3-EMB2-08 are identified as **Confirmatory Item CI-SRP3.9.3-EMB2-08**.

The AP1000 component design has been completed to the extent that the COL Information Item in Revision 15 is not necessary, allowing the aspects of the COL Information Item addressing components to be eliminated, as documented in the amendment to the DCD.

Since its original Design Certification (DC) amendment, Westinghouse has modified the AP1000 seismic design ground motion requirements, in order to extend the DC application to soil sites. It was expected that these revised seismic loadings would have an impact on the component designs already performed up to that point. In RAI-SRP3.9.3-EMB2-02, the staff requested Westinghouse to confirm that, for all the major ASME Code Section III, components already designed, all the pertinent design specifications and design reports have been updated to incorporate the effects of the newly modified seismic loadings. By letter dated June 28, 2008,

Westinghouse changed the design basis for the major ASME Code, Section III components to include the design spectra and seismic requirements envelope for the hard rock and associated with the expanded soil conditions (six soils case). These revised seismic design requirements for the six soils case are included in the design specifications for the major ASME components. Westinghouse stated the analyses supporting the as-designed design reports prepared or being completed for NRC review were in compliance with the design specifications and include these revised seismic requirements of the six soils case. Based on the above response and the confirmation obtained from the staff's onsite review, the staff found Westinghouse has adequately incorporated the latest revised seismic input motions for the component design. RAI-SRP3.9.3-EMB2-02 is, therefore, closed.

In APP-GW-GLR-115, Revision 2, Section 6.2.2, Tables 6.2.2-2, -3, -5 provided the resultant forces of three sample components of the primary coolant loop. The analysis of these components were to evaluate the comparison of the loads from the hard rock high frequency (HRHF) input to those obtained from the time history associated with the Certified Seismic Design Response Spectra (CSDRS) input. For the three sample components, the applicant indicated the resultant forces due to CSDRS (Certified Seismic Design Response Spectra) excitation are higher than the resultant forces caused by the HRHF excitation. Therefore, the HRHF loads will not govern the component design, but the CSDRS loads will. The staff found the loads of HRHF excitation and seismic loads to components were adequately incorporated in Revision 2 of the APP-GW-GLR-115 report and are acceptable.

The staff's review of Westinghouse's evaluation on the effects of high frequency seismic input on the AP1000 mechanical component design is provided in Section 3.10 of this report.

3.9.3.3 Conclusions

Based on the information provided in the Westinghouse's responses to the RAIs, the staff finds, with the exception of confirmatory items, the application meets the guidance of RG 1.206, SRP 3.9.3 and the regulations for the design of mechanical components, and is acceptable. Piping-related issues are discussed in Section 3.12 of this report.

3.9.4 Control Rod Drive Systems

In Revision 17 to the AP1000 DCD, Westinghouse proposed changes to the hydrostatic test pressure for the CRDM housing as well as other materials related to changes to the CRDM. This resulted in changes to the DCD in Sections 3.5.1.2.1.1, 3.9.4.1.1, 3.9.4.3, and 4.1.1. By a letter dated November 15, 2006, Westinghouse submitted AP1000 COL TR-30, "AP1000 CRDM Design," APP-GW-GLN-013, Revision 0 to provide the technical justification for the proposed changes.

As stated in Revision 15 to the AP1000 DCD, Sections 3.5.1.2.1.1, 3.9.4.1.1, and 4.1.1, the specified hydrostatic test pressure for the CRDM is 150 percent of the system design pressure. In Section 3.9.4.1.1 of the DCD, the attachment of the latch assembly housing is described as a shrink-fit and partial penetration weld of the latch assembly housing. However, the latch assembly housing will be welded to the CRDM nozzle by a bi-metallic weld. Also, Section 3.5.1.2.1.1 describes the attachment of the latch assembly housing to a head adapter when in fact the latch assembly housing will be welded to an Alloy 690 nozzle. In Revision 17 to the

DCD, Westinghouse proposed to hydrostatically test the CRDM at 125 percent of system design pressure and to describe the correct fabrication sequence and terminology for the assembly.

In addition, in Section 3.9.4.1.2, "Control Rod Withdrawal," 3.9.4.1.3, "Control Rod Insertion" and 3.9.4.1.4, "Holding and Tripping of the Control Rods" the applicant proposed modifications to the sequence of events for withdrawal, insertion, holding and tripping of control rods.

The DCD was modified to clarify the classification of the CRDM latch assembly, the CRDM drive rod assembly, CRDM coil stack assembly, and the CRDM position indicator.

3.9.4.1 Evaluation

3.9.4.1.1 Hydrostatic Testing and Attachment of the Latch Assembly Housing

Westinghouse revised AP1000 DCD, Sections 3.5.1.2.1.1, 3.9.4.1.1, and 4.1.1 to reduce the hydrostatic test pressure for the CRDM from 150 percent to 125 percent of system design pressure. The stated reason for this change was the requirements of ASME Boiler and Pressure Vessel Code (ASME Code). Section III, Paragraph NB-6221 specifies that nuclear power plant components are tested at 125 percent of system design pressure. The staff finds the proposed change acceptable because the proposed hydrostatic test pressure of 125 percent of system design pressure meets the requirements of ASME Code, Section III, which the NRC staff had incorporated by reference in 10 CFR 50.55a, "Codes and Standards."

AP1000 DCD, Revision 15, Section 3.9.4.1.1, states the attachment of the latch assembly housing to the vessel head is accomplished by a shrink-fit and partial penetration weld. Westinghouse determined the latch housing will be welded to the Alloy 690 nozzle with a bi-metallic weld and the nozzle will be attached to the reactor vessel head by a shrink-fit and partial penetration weld. In Revision 17 to the DCD, Westinghouse revised Sections 3.5.1.2.1.1, 3.9.4.1.1, and 3.9.4.3 to describe the correct fabrication sequence and correct terminology for these components. The staff finds the proposed changes are an editorial change to the AP1000 DCD and as such do not affect the design basis of the component. Furthermore, the proposed changes describe the correct fabrication sequence, uses the correct terminology and, therefore, are acceptable.

The NRC staff reviewed the proposed changes as they relate to Revision 17 to the AP1000 DCD. The proposed changes, as identified in TR-30, have been adequately incorporated into Revision 17 to the DCD. Accordingly, these changes are generic and are expected to be used by all COL applications referencing the AP1000 certified design.

3.9.4.1.2 Control Rod Sequence of Events

In Section 3.9.4.1.2 "Control Rod Withdraw," Section 3.9.4.1.3 "Control Rod Insertion" and 3.9.4.1.4, "Holding and Tripping of the Control Rods," the applicant proposed to modify the control rod withdrawal and insertion sequence order. Specifically, during control rod withdrawal the moveable gripper coil B is in the de-energized ("OFF") state instead of the energized ("ON") state. Furthermore, insertion of control rods initiates with the moveable gripper coil B in the de-energized ("OFF") state instead of the lift coil C in the energized ("ON") state.

The applicant proposed to change the DCD in Section 3.9.4.1.4 "Holding and Tripping of the Control Rods" to be in accord with the proposed change in Section 3.9.4.1.1 "Control Rod Drive Mechanism (CRDM)." The proposed change reiterates that in the holding mode both the stationary gripper coil A and the moveable gripper coil B are energized. Additionally the applicant elaborates the drive rod assembly is held in position by three latches on the stationary gripper and three latches on the moveable gripper. As a result of the proposed modification, the applicant clarifies a reactor trip occurs when power to the stationary as well as to the moveable gripper coils is cut off.

The staff finds the proposed changes to the sequence of events for control rod withdrawal, control rod insertion, and holding and tripping the control rod do not adversely affect the ability of the AP1000 CRDM to perform its safety-related functions.

3.9.4.1.3 Seismic Qualification of CRDM

The staff became aware of discussions internationally concerning the classification and qualification of the CRDM latch assembly. Based on these discussions, the staff determined the seismic qualification of the CRDM standard design may not meet requirements. GDC 2 of Appendix A to 10 CFR Part 50 states structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, combined with appropriate effects of normal and accident conditions, without loss of capability to perform their safety functions.

In RAI-SRP3.9.4-EMB1-01, the staff requested Westinghouse to provide a justification to explain why the latch mechanism and coil stack assembly do not need to be seismically qualified to comply with GDC 2, or to revise the seismic classifications of the CRDM components to ensure adequate seismic qualification for the safety functions of the Control Rod Drive System. In RAI-SRP3.9.4-EMB1-02, the staff requested further clarification on design changes discussed internationally.

Westinghouse provided justification for the equipment classification for the latch assembly and coil stack assembly and why they do not need to be seismically qualified. The justification is based on 1) the design finality of the AP1000 Design Certification, 2) the precedent of operating plants, and 3) the function of the latch assembly and coil stack assembly in the AP1000 CRDM.

The staff finds Westinghouse's justification for not qualifying the latch assembly using the precedent of operating plants and any postulated failure of the latch assembly results in a dropped rod and a subsequent increase in negative reactivity (justifications 2 and 3 above) unacceptable. Operating plants were licensed using a lower required response spectra (RRS) compared to new reactors. The RRS for new reactors is much higher. Jamming of the latch mechanism is a postulated failure which results in no dropped rod and subsequently no reactivity change. Further, in the response to RAI-SRP3.9.4-EMB1-02, Westinghouse indicated that although international approaches to safety classification and requirements for safety class equipment may differ from the NRC requirements, it expects the design, fabrication and quality assurance requirements for the CRDM latch assemblies will remain common with the requirements for latch assemblies manufactured for U.S. applications.

In its response, Westinghouse referred to discussion in Chapter 15. There are only three postulated events that assume credit for reactivity control systems, other than a reactor trip to render the plant subcritical. These events are the steam-line break, feedwater-line break, and small break loss of coolant accident. The reactivity control systems in these accidents are the reactor trip system and the passive core cooling system (PXS). The probability of a common mode failure impairing the ability of the reactor trip system to perform its safety-related function is extremely low. However, analyses performed to demonstrate compliance with the requirements of 10 CFR 50.62 demonstrate safety criteria would not be exceeded even if the control rod drive system were rendered incapable of functioning during anticipated transients for which its function would normally be expected. The evaluation demonstrates borated water from the core makeup tank shuts down the reactor with no rods required, and the passive residual heat removal system provides sufficient core heat removal. Due to these additional safety measures, Westinghouse concluded the latch assembly and all other active mechanical components of the CRDM are not required to be classified as safety-related.

Based on finality, the low probability of common mode failure, and the argument that existing additional safety measures limit the safety consequence, Westinghouse has provided adequate justification to maintain the current classifications for the latch mechanism and coil stack assembly. Additionally, Westinghouse does not expect any changes to the design, fabrication and quality assurance requirements for the CRDM latch assemblies. The staff finds the responses to RAI-SRP3.9.4-EMB1-01 and RAI-SRP3.9.4-EMB1-02 acceptable and RAI-SRP 3.9.4-EMB1-02 is closed.

As a result of RAI-SRP3.9.4-EMB1-01, Westinghouse proposed modifications to DCD Tier 2 Table 3.2-3 to clarify the classification of the CRDM latch assembly, the CRDM drive rod assembly, CRDM coil stack assembly, and the CRDM position indicator. Additionally, DCD Tier 2 Section SR 3.1.4.3 (of the Chapter 16 Technical Specifications) will be modified to include drop tests after each earthquake requiring shutdown.

The staff finds these proposed revisions acceptable and this response to RAI-SRP3.9.4-EMB1-01 will be **Confirmatory Item CI-SRP3.9.4-EMB1-01**.

3.9.4.2 Conclusion

The staff further concludes the applicant's proposed changes do not adversely affect the ability of the AP1000 CRDM to perform its safety-related functions. On the basis that the AP1000 control rod drive system design continues to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds the changes to the CRDM design description provided in AP1000 DCD, Revision 17, are acceptable.

3.9.5 Reactor Pressure Vessel Internals

In Revision 16 of the AP1000 DCD, the applicant added two new components (neutron panels to Subsection 3.9.5.1.1, and a flow skirt to Subsection 3.9.5.1.4) to the design of the reactor vessel internal structure. The subsequent Revision 17 of DCD Section 3.9.5 included minor changes to incorporate responses to the staff's RAIs for DCD Revision 16. DCD Revision 17 did not propose any additional new core support structure or reactor internals components requiring further technical evaluation.

3.9.5.1 Evaluation

The staff reviewed the proposed changes to the reactor vessel internals in the AP1000 DCD Revision 17 in accordance with the guidance in the SRP Section 3.9.5, "Reactor Pressure Vessel Internals." The regulatory basis for Section 3.9.5 of the AP1000 DCD is documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The following evaluation discusses the results of the staff's review.

3.9.5.1.1 Neutron Panels

In response to RAI-SRP3.9.5-EMB1-04 received in a letter dated June 20, 2008, the applicant stated the function of the neutron panels is to protect the reactor vessel from detrimental radiation effects by limiting the total exposure in the localized regions of the vessel wall in closest proximity to the core outer boundaries. The applicant also clarified the neutron panels are classified as internal structures, and, for conservatism, the neutron panels are analyzed in accordance with the requirements of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code, Section III (ASME III), Subsection NG. The neutron panels are fabricated from material complying with ASME III, NG-2000 and are designed and analyzed per ASME III, NG-3000. The neutron panels are attached to the core barrel with threaded fasteners. The applicant also stated that the neutron panels have been sized to prevent excessive thermal loading on the bolts and to withstand flow, thermal and vibratory loading. In addition the bolts and preload of the bolts have been sized to accommodate radiation relaxation and radiation induced gamma heating such that the preload is maintained. These bolts are secured by locking devices. Oscillatory forces on the neutron panels have been calculated based on the turbulence in the annulus between the neutron panels and the reactor vessel based on the correlation with past scale model tests and computational fluid dynamics (CFD) analysis. The analysis of the forces, as discussed was evaluated to assure the preload is maintained and design limits are achieved. The applicant also stated the AP1000 reactor vessel inside diameter has been increased by two inches over the core elevations where the neutron panels were added. This results in a net flow area increase of 4 percent in the downcomer relative to the flow area before the panels were added. Thus, the lower average downcomer velocity is expected to mitigate the potential for any adverse effects of flow-induced vibration caused by the added neutron panels.

3.9.5.1.2 Flow Skirt

The flow skirt is a perforated cylindrical ring structure attached to the reactor vessel bottom head at an elevation just below the lower core support plate. The flow skirt provides a more uniform distribution of inlet flow from the reactor vessel downcomer annulus to the core inlet nozzles in the lower core support plate. Although the flow skirt is welded to the reactor vessel, since the structure is located entirely within the pressure boundary, it is treated in the DCD as a reactor vessel internal structure. In response to RAI-SRP3.9.5-EMB1-01 received in a letter dated June 20, 2008, the applicant clarified although classified as an internal structure (as opposed to a core support structure), for conservatism the flow skirt is analyzed in accordance with the requirements of ASME III, Subsection NG. The ASME Code jurisdictional boundary requires the attachment weld between the flow skirt and the reactor vessel flow skirt support lug to be designed and analyzed to ASME Section III, Subsection NB-3200. All other design details

of the flow skirt conform to ASME III, Subsection NG-3000 requirements. The applicant also stated the flow skirt design includes flow-induced vibratory loading considerations including downcomer flow turbulence, random turbulence within the reactor vessel lower head, and vortex shedding through the flow skirt perforations. The flow skirt design specification requires the structural design qualification calculations for the flow skirt meet the requirements of ASME III, Subsection NG-3000.

In response to RAI-SRP3.9.5-EMB1-02, dated June 20, 2008, the applicant stated the primary function of the flow skirt is to assure the distribution of flow entering the core is within prescribed limits for fuel assembly inlet flow mismatch. A CFD analysis of a reactor vessel/internals model, which included the inlet nozzle, downcomer, lower plenum (including secondary core support and vortex suppression structures), and lower core support plate, was performed by the applicant to determine the core inlet flow distribution. The Computational Fluid Dynamics (CFD) approach used in the analyses was used for analyses of similar operating reactor vessel internals geometry, and was benchmarked to scale model testing data with good agreement. The applicant performed analyses both with and without a flow skirt. Without the flow skirt the limits for uniformity of core inlet flow distribution were not met. In response to RAI-SRP3.9.5-EMB1-03, dated June 20, 2008, the applicant provided a figure of the flow skirt which clarified its form and function, and the staff considers this RAI closed.

3.9.5.1.3 Component Classification and Design Basis

The neutron panels are also classified as reactor internal structures (as opposed to core support structures), and, for conservatism, are designed according to the requirements of ASME III, Subsection NG-3000, even though the ASME III Code requires this approach only for internals components classified as core support structures. The flow skirt is also designed per the requirements of ASME III, Subsection NG. As provided by the ASME Code, Section III, Subsection NG-1122(b) and (c), these internal structure components must be constructed so as not to adversely affect the integrity of the core support structures, but the specific design requirements of ASME III, Subsection NG are not required unless so stipulated by the designer. The applicant has conservatively chosen to use the requirements of Subsection NG-3000 for the design of both the flow skirt and the neutron panels.

The staff conducted a design audit at the Westinghouse Energy Center in Monroeville, PA during October 13-17, 2008 (Reference 1). The audit included review of the ASME III Code design documentation for the AP1000 reactor pressure vessel and the reactor internals and core support structure. The results of this audit are contained in NRC letter dated December 30, 2008, Docket No. 52-006, Subject: Summary of the October 13-17, 2008, On-site Review of the AP1000 Component Design. The staff confirmed the neutron panels are part of the reactor internals design specification and design report, and the flow skirt has its separate design specification and analysis report. The audit verified the design bases for the neutron panels and flow skirt incorporate the requirements of ASME III, Subsection NG-3000. The design analyses for the neutron panels show the results meet the design margins required by ASME III, Subsection NG. Although the design report analysis for the flow skirt was not complete at the time of the audit, the flow skirt design specification clearly established the design requirements according to the provisions of ASME III, Subsection NG. Therefore, the staff concluded the design methodology meets the review criteria of SRP Section 3.9.5, and is acceptable.

In DCD Subsection 3.9.2.3, the applicant stated the results of the Trojan 1 reactor tests showed the lower internals vibrations are lower with neutron panels than with a circular thermal shield. Additionally, as stated above, a net flow area increase of 4 percent in the AP1000 downcomer relative to the flow area before the neutron panels were added results in a lower average flow velocity in the downcomer annulus. The lower average downcomer flow velocity will tend to mitigate the potential effects of any localized turbulence added by the neutron panels. On this basis, the staff concluded there is reasonable assurance the added neutron panels will not be adversely affected by flow-induced vibration (FIV).

As indicated above, the applicant considered the flow-induced vibratory loading including downcomer flow turbulence and random turbulence for the flow skirt. The structural qualification requirements for the flow skirt and the neutron panels are consistent with the provisions of ASME III, Subsection NG. The applicant's CFD analyses used for prediction of flow-induced vibratory loading coupled with pre-operational FIV testing (as discussed in Section 3.9.2.3 of NUREG-1793) will ensure there are no adverse effects of FIV and flow-excited acoustic resonances on the reactor vessel internal structures. On this basis, the staff finds the flow skirt and neutron panels will not cause adverse flow effects within the reactor vessel internal structures during normal operation or anticipated operational transients.

3.9.5.2 Conclusion

The applicant has met the regulatory requirements of GDC 1 and 10 CFR 50.55a by designing the neutron panels and the flow skirt to quality standards commensurate with the importance of the safety functions performed. The design criteria used for these two newly added reactor internals components are in compliance with the requirements of the 1998 Edition, including 1999 and 2000 Addenda, of ASME III, Subsection NG-3000.

The applicant has met the regulatory requirements of GDCs 2, 4, and 10 by designing these reactors internals components to withstand the effects of normal operation and postulated accident loadings with sufficient margin to maintain their structural integrity to assure they do not adversely affect the integrity of the safety-related reactor core support structures. The applicant has also designed these reactor internals components to assure acceptable fuel design and performance limits are met during conditions of normal operation and anticipated operational occurrences.

The staff concludes the design bases for the neutron panels and for the flow skirt meet the staff review criteria of SRP 3.9.5, including the regulatory requirements of 10 CFR 50.55a, GDCs 1, 2, 4, and 10, and are, therefore, acceptable.

3.9.6 Testing of Pumps and Valves

In Revision 17 to the AP1000 DCD Tier 2, Westinghouse modified Section 3.9.6, "Inservice Testing of Pumps and Valves," including Table 3.9-16, "Valve Inservice Test Requirements." Westinghouse incorporated changes to AP1000 DCD Tier 2, Section 3.9.6, to support the description of the Inservice Testing (IST) Program required to be provided by a COL applicant.

3.9.6.1 Evaluation

In Section 3.9.6, "Testing of Pumps and Valves," of NUREG-1793, the NRC staff described its review of the description of the IST Program for the AP1000 design provided in AP1000 DCD Tier 2, Section 3.9.6. Other sections of the AP1000 DCD addressed the design of safetyrelated valves, and inservice inspection and testing of dynamic restraints. As discussed in NUREG-1793, the development of a complete plant-specific IST Program falls outside the scope of Design Certification. At the Design Certification stage, it is necessary to establish a baseline Code edition and addenda to ensure the IST requirements of the baseline ASME Code can be performed without exception, and that the design of the AP1000 systems and components provides access to permit the performance of testing pursuant to the NRC regulations specified in 10 CFR Part 50, Section 50.55a.

AP1000 DCD Tier 2, Section 3.9.6 states inservice testing of ASME *Boiler & Pressure Vessel Code* (BPV Code), Section III, Class 1, 2 and 3 pumps and valves is performed in accordance with the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) and applicable addenda, as required by 10 CFR Part 50, Section 50.55a(f), except where specific relief has been granted by the NRC. The baseline ASME OM Code used to develop the IST plan for the AP1000 Design Certification was the 1995 Edition and 1996 Addenda. AP1000 DCD Tier 2, Section 3.9.6 provides a general description of the IST Program to be developed for the AP1000 reactor to satisfy the requirements in 10 CFR Part 50, Section 50.55a and the provisions of the ASME OM Code incorporated by reference in the NRC regulations. In NUREG-1793, the NRC staff found the IST Program description in the AP1000 DCD to be acceptable for the AP1000 Design Certification, and that the AP1000 DCD had not taken exception to any ASME OM Code requirements established in the 1995 Edition and 1996 Addenda.

Since the issuance of NUREG-1793, the NRC has determined a COL applicant referencing the AP1000 design needs to fully describe the IST, Motor-Operated Valve (MOV) Testing and other operational programs as defined in Commission Paper SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria." RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," provides guidance for a COL applicant in preparing and submitting its COL application in accordance with the NRC regulations. For example, Section C.IV.4 in RG 1.206 discusses the requirement in 10 CFR 52.79(a) for descriptions of operational programs need to be included in the Final Safety Analysis Report (FSAR) in a COL application to support a reasonable assurance finding of acceptability. A COL applicant may rely on information in the applicable Design Certification to help provide a full description of the operational programs for the COL application. At a public meeting on March 26 and 27, 2008, Westinghouse indicated the AP1000 DCD will address issues common to COL applicants implementing the AP1000 design. Therefore, the NRC staff reviewed the revision to the AP1000 DCD related to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints, including DCD provisions intended to minimize the supplemental information necessary to be provided by a COL applicant in fully describing the operational programs in support of its COL application for an AP1000 reactor. As described below, the NRC staff concludes the revision to Section 3.9.6 of the AP1000 DCD continues to provide an acceptable description of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints sufficient for the AP1000 Design Certification in accordance with the NRC regulations and the ASME Code requirements incorporated by reference in the NRC regulations, with provisions for the consideration of

lessons learned from nuclear power plant operating experience, pending resolution of the identified confirmatory items in this section.

A COL applicant may reference the provisions in Section 3.9.6 to the AP1000 DCD as part of its responsibility to fully describe the IST, MOV Testing, and other operational programs in support of its COL application. AP1000 DCD Tier 2, Section 3.9.6 states Table 3.9-16, "Valve Inservice Test Requirements," identifies the components subject to the preservice and IST programs, and the method and frequency of pre-service and inservice testing. The NRC staff will evaluate the full description of the IST Program provided by a COL applicant during review of the COL application consistent with RG 1.206 and NRC Standard Review Plan (SRP) Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," provides guidance for the preparation of IST Program documentation and tables. Following COL issuance, the NRC staff will evaluate development and implementation of the IST Program prior to and during plant operation.

AP1000 DCD Tier 2, Subsection 3.9.6.1, "Inservice Testing of Pumps," specifies the AP1000 reactor design does not include pumps with safety functions with the exception of the coastdown of the reactor coolant pumps. The proposed changes to the AP1000 DCD do not affect the use of pumps with respect to safety-related applications. Therefore, the IST Program described in the proposed revision to the AP1000 DCD does not include pumps. As determined in NUREG-1793, the NRC staff considers the IST Program scope for the AP1000 design with respect to pumps to be acceptable.

AP1000 DCD Tier 2 discusses the functional design and qualification of safety-related valves and dynamic restraints in several sections. For example, Subsection 3.9.3.2, "Pump and Valve Operability Assurance," in AP1000 DCD Tier 2, Chapter 3, "Design of Structures, Components, Equipment and Systems," refers to operational tests to verify the valve opens and closes prior to installation. AP1000 DCD Tier 2, Subsection 3.9.3.2.2 specifies cold hydro tests, hot functional tests, periodic inservice inspections, and periodic inservice operations to be performed in situ to verify the functional capability of the valves. Section 5.4.8, "Valves," of Section 5.4, "Component and Subsystem Design," in AP1000 DCD Tier 2, Chapter 5, "Reactor Coolant System and Connected Systems," includes provisions regarding design and qualification, and preoperational testing of valves within the scope of Chapter 5, and refers to these activities for other safety-related valves. AP1000 DCD Tier 2, Subsection 5.4.8.3, "Design Evaluation," states the requirements for qualification testing of power-operated active valves are based on ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," as listed in AP1000 DCD Tier 2, Section 5.4.16, "References." AP1000 DCD Tier 2, Section 5.4.9, "Reactor Coolant System Pressure Relief Devices," includes provisions for design, testing, and inspection of relief devices in the reactor coolant system. AP1000 DCD Tier 2, Section 5.4.10, "Component Supports," includes provisions for design, testing, and inspection of component supports in the reactor coolant system. During the public meeting on March 26 and 27, 2008, Westinghouse discussed its development of design and procurement specifications for safety-related valves and dynamic restraints for the AP1000 reactor design. In RAI-SRP3.9.6-CIB1-01, the NRC staff requested that Westinghouse provide a schedule for the availability of the design and procurement specifications for safety-related valves and dynamic restraints to be used in the AP1000 reactor for NRC staff review. In its response to this RAI, in a letter dated July 18, 2008, Westinghouse

reported the design and procurement specifications would be made available for NRC review.

On October 14 and 15, 2008, the NRC staff conducted an audit of design and procurement specifications for pumps, valves, and dynamic restraints to be used for the AP1000 reactor at the Westinghouse office in Monroeville, Pennsylvania. The staff found Westinghouse had included ASME Standard QME-1-2007 in its design and procurement specifications for AP1000 components. ASME QME-1-2007 incorporates lessons learned from valve testing and research programs performed by the nuclear industry and NRC Office of Nuclear Regulatory Research. In a memorandum dated November 6, 2008, the NRC staff documented the results of the audit with the specific OIs (ADAMS Accession Number ML083110154). The audit response was tracked as OI-SRP3.9.6-CIB1-01. In a letter dated January 26, 2010, Westinghouse provided its planned response to the audit follow-up items. First, Westinghouse stated a reference to ASME QME-1-2007 will be included in AP1000 DCD Tier 2, Section 3.9. Second, Westinghouse stated the basis for the assumptions for valve seat coefficients of friction for gate and globe valves is derived from the Joint Owners Group (JOG) Program on MOV Periodic Verification as a starting point for the initial actuator sizing. Westinghouse indicated the final basis for the friction coefficient values will be derived in accordance with an approved methodology contained in ASME QME-1-2007. Third, Westinghouse stated the applicable valve design specification indicates active valves must be qualified in accordance with the ASME QME-1 standard, and the specification will be further clarified to indicate any existing testing used to demonstrate functional qualification must fully satisfy the provisions of ASME QME-1-2007. Fourth, Westinghouse stated the AP1000 DCD Tier 2, Figure 6.3-1, "Passive Core Cooling System Piping and Instrumentation Diagram," will be revised to include test connections to allow flow testing of Core Makeup Tank Discharge Check Valves PXS-V016A/B and V017A/B in both the forward and reverse directions. In September 2009, the NRC issued Revision 3 to RG 1.100, "Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," which accepts the use of ASME QME-1-2007, with certain staff positions, for the functional design and qualification of safety-related pumps, valves, and dynamic restraints.

On May 17, 2010, the NRC staff conducted a follow-up audit at the Westinghouse office in Rockville, Maryland, to review the revisions to the design and procurement specifications prepared since the October 2008 audit. The staff conducted telephone conferences on May 19 and 28, and June 10, 2010, to support close-out of the audit. Based on the follow-up audit, the NRC staff found Westinghouse has updated the design and procurement specifications to address NRC comments provided during the October 2008 audit. For example, the staff found the design and procurement specifications require the application of ASME Standard QME-1-2007 for the qualification of mechanical equipment to be used in an AP1000 reactor. The design and procurement specifications also have been revised to incorporate additional comments provided by the staff during the October 2008 audit. In a memorandum dated June 15, 2010, the NRC staff documented the results of the follow-up audit (ADAMS Accession Number ML101650024). Based on the results of the follow-up audit, OI-SRP3.9.6-CIB1-01 is closed. **Confirmatory Item CI-SRP3.9.6-CIB1-01** will be used to track the planned AP1000 DCD Tier 2 changes in Sections 3.9 and 6.3.

AP1000 DCD Tier 2, Section 3.9.2, "Dynamic Testing and Analysis," describes tests to confirm

piping, components, restraints, and supports have been designed to withstand the dynamic effects of steady-state FIV and anticipated operational transient conditions. Subsection 14.2.9.1.7, "Expansion, Vibration and Dynamic Effects Testing," in AP1000 DCD Tier 2, Chapter 14, "Initial Test Program," states the purpose of the expansion, vibration and dynamic effects testing is to verify the safety-related, high energy piping and components are properly installed and supported such that, in addition to other factors, vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue to safety-related plant systems. Nuclear power plant operating experience has revealed the potential for adverse flow effects from vibration caused by hydrodynamic loads and acoustic resonance on reactor coolant, steam, and feedwater systems. As part of the functional design and qualification for AP1000 components, the COL applicant will be responsible for addressing the provisions in the AP1000 DCD for consideration of potential adverse flow effects on safety related valves and dynamic restraints within the IST Program in the reactor coolant, steam, and feedwater systems from hydraulic loading and acoustic resonance during plant operation.

AP1000 DCD Tier 2, Subsection 3.9.6.2, "Inservice Testing of Valves," refers to the use of nonintrusive techniques to periodically assess degradation and performance of selected valves. In RAI-SRP3.9.6-CIB1-02, the NRC staff requested that Westinghouse to clarify the use of nonintrusive techniques within the IST Program to support implementation of this subsection by a COL applicant referencing the AP1000 reactor design. In its response to this RAI, in a letter dated September 9, 2008 (ADAMS Accession Number ML082560239), Westinghouse stated it will be the responsibility of the licensee to define the nonintrusive technique and methods for periodic assessment of check valve performance and degradation. Also in response to this RAI, Westinghouse modified Subsection 3.9.6.2 in Revision 17 to the AP1000 DCD Tier 2 to state inservice testing may incorporate the use of nonintrusive techniques to periodically assess degradation and performance of selected check valves. The NRC staff finds the Westinghouse response to this RAI and Revision 17 to the AP1000 DCD clarify the use of nonintrusive techniques referenced in the AP1000 DCD, and the COL licensee will define any nonintrusive techniques will be implemented. Therefore, RAI-SRP3.9.6-CIB1-02 is closed.

The revision to AP1000 DCD Tier 2, Subsection 3.9.6.2 specifies testing of power-operated valves (POVs) used in the AP1000 reactor will utilize guidance from Generic Letter (GL) 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," and the JOG MOV Periodic Verification Program. The NRC staff accepted the JOG Program on MOV Periodic Verification as an industry-wide response to GL 96-05 for valve age-related degradation in a safety evaluation dated September 25, 2006 (ADAMS Accession Number ML061280315) with a supplement dated September 18, 2008 (ADAMS Accession Number ML082480638). In RAI-SRP3.9.6-CIB1-03, the NRC staff requested that Westinghouse describe the incorporation of lessons learned from valve programs in planning the IST Program for POVs other than MOVs to support implementation of this subsection by a COL applicant referencing the AP1000 reactor design. In its response to this RAI, in a letter dated September 9, 2008, Westinghouse stated AP1000 DCD Tier 2, Subsection 3.9.6.2 would be revised to address this RAI. As a result, Revision 17 to the AP1000 DCD Tier 2, Subsection 3.9.6.2, states guidance from applicable NRC generic letters and industry guidelines is reflected in the IST provisions in AP1000 DCD Tier 2, Table 3.9-16. Revision 17 to the AP1000 DCD also specifies lessons learned from GL 96-05 and the JOG MOV Periodic Verification Program are reflected in the IST Program and valve procurement testing

requirements. Revision 17 to the AP1000 DCD indicates the IST Program requires periodic updating that takes into account changes to the diagnostic methods and test equipment, emergent industry issues, and equipment alignment. The NRC staff finds the Westinghouse response to RAI-SRP3.9.6-CIB1-03 and the provisions specified in Revision 17 to the AP1000 DCD provide an acceptable clarification as part of the AP1000 Design Certification that the lessons learned from valve operating experience and testing programs will be included in the IST and procurement programs for AP1000 nuclear power plants. RAISRP3.9.6-CIB1-03 is closed.

AP1000 DCD Tier 2, Subsection 3.9.6.2 states the operability test for safety-related POVs with an active function may be either a static or a dynamic (flow and differential pressure) test. In RAI-SRP3.9.6-CIB1-04, the NRC staff requested that Westinghouse clarify the use of static tests for operability determinations of POVs to support implementation of this subsection by a COL applicant referencing the AP1000 reactor design. In its response to this RAI, in a letter dated September 9, 2008, Westinghouse stated AP1000 DCD Tier 2, Subsection 3.9.6.2 would be revised to address this RAI. As a result, Revision 17 to the AP1000 DCD Tier 2, Subsection 3.9.6.2 references Subsection 3.9.6.2.2 for the use of static or dynamic testing for safety-related POVs. The NRC staff considers this clarification of Subsection 3.9.6.2 to be sufficient to close this RAI, but the use of static or dynamic testing for safety-related POVs will be addressed as part of RAI-SRP3.9.6-CIB1-08 discussed later in this safety evaluation. RAI-SRP3.9.6-CIB1-04 is closed.

The revision to AP1000 DCD Tier 2, Subsection 3.9.6.2.2 states the frequency for a position indication test will be once every 2 years unless otherwise justified. In RAI-SRP3.9.6-CIB1-07, the NRC staff requested that Westinghouse clarify the need for a COL applicant to request relief from or an alternative to the ASME OM Code testing requirement with respect to position indication if the Code provisions are not satisfied. In its response to this RAI in a letter dated July 14, 2008 (ADAMS Accession Number ML081980186), Westinghouse noted that AP1000 valves that require position indication testing, as documented in AP1000 DCD Tier 2, Table 3.9-16, are identified as having a 2 year frequency. Westinghouse indicated no relief is requested for position indication testing. The NRC staff considers the position indication testing frequency in the AP1000 DCD to be consistent with the ASME OM Code. The COL applicant will need to request relief from, or an alternative to, the ASME OM Code provisions if the position indication testing frequency will not be satisfied. RAI-SRP3.9.6-CIB1-07 is resolved.

AP1000 DCD Tier 2, Subsection 3.9.6.2.2 discusses POV testing in a subsection titled "Power-Operated Valve Operability Tests." The revision to the AP1000 DCD specifies operability testing as required by 10 CFR Part 50, Section 50.55a(b)(3)(ii) is performed on MOVs that are included in the ASME OM Code IST Program to demonstrate the MOVs are capable of performing their design-basis safety functions. In RAI-SRP3.9.6-CIB1-08, the NRC staff requested that Westinghouse to clarify the discussion of POV operability testing in the AP1000 DCD to support implementation of the DCD provisions by a COL applicant referencing the AP1000 reactor design. In response to this RAI, in a letter dated September 9, 2008, Westinghouse described planned changes to AP1000 DCD Tier 2, Subsection 3.9.6.2.2 to address this RAI. The NRC staff determined RAI-SRP3.9.6-CIB1-08 needed to remain open until several aspects of the planned AP1000 DCD changes were clarified as discussed below for OI-SER3.9.6-CIB1-02, 03, 04, and 05.

As OI-SRP3.9.6-CIB1-02, the NRC staff tracked the need for the reference to static testing of valves in the AP1000 DCD to be consistent with the JOG MOV Periodic Verification Program, which might require dynamic testing based on the results of the evaluation of the MOV margin. In letters dated January 26, February 18, and March 5, 2010, Westinghouse provided planned changes to the AP1000 DCD Tier 2, Section 3.9.6 to specify POV testing will be consistent with the JOG MOV Periodic Verification Program, and removed the reference to static-only testing. Westinghouse also removed the discussion of testing of POVs outside the scope of the JOG MOV Periodic Verification Program because safety-related MOVs to be used at AP1000 plants will be within the scope of the JOG MOV Periodic Verification Program. The staff considers these planned DCD changes will resolve this portion of RAI-SRP3.9.6-CIB1-08. Therefore, OI-SRP3.9.6-CIB1-02 is closed. The planned changes to the AP1000 DCD will be tracked as **Confirmatory Item CI-SRP-3.9.6-CIB1-02**.

As OI-SRP3.9.6-CIB1-03, the NRC staff tracked the need for the AP1000 DCD to specify the edition of the ASME Standard QME-1 referenced in Section 3.9 because the NRC staff has not accepted ASME Standard QME-1 editions issued prior to 2007 as an acceptable functional qualification approach for valves. In letters dated January 26 and February 18, 2010, Westinghouse indicated AP1000 DCD Tier 2, Section 3.9 would be revised to reference ASME QME-1-2007. The staff considers this planned change to the AP1000 DCD will resolve this portion of RAI-SRP3.9.6-CIB1-08. Therefore, OI-SRP3.9.6-CIB-03 is closed. The planned changes to the AP1000 DCD will be tracked as **Confirmatory Item CI-SRP-3.9.6-CIB1-03**.

As OI-SRP3.9.6-CIB1-04, the NRC staff tracked the need for the planned application of ASME OM Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants," within the AP1000 IST Program to be consistent with the edition of Code Case OMN-1 accepted in RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," or to indicate the need for the submission of a request to implement an alternative to the OM Code. In letters dated January 26, February 18, and March 5, 2010, Westinghouse provided a planned revision to AP1000 DCD Tier 2, Section 3.9.6 that will specify use of ASME OM Code Cases must be consistent with RG 1.192. The NRC staff considers this planned revision to the AP1000 DCD to be acceptable. A COL applicant or licensee planning to use an ASME OM Code Case not accepted in RG 1.192 will need to submit a request to implement an alternative to ASME OM Code as required by 10 CFR 50.55a. The staff considers this planned change to the AP1000 DCD will resolve this portion of RAI-SRP3.9.6-CIB1-08. Therefore, OISRP3.9.6-CIB1-04 is closed. The planned changes to the AP1000 DCD will be tracked as **Confirmatory Item CI-SRP-3.9.6-CIB1-04**.

As OI-SRP3.9.6-CIB1-05, the NRC staff tracked the need for the Technical Specifications and Technical Specification Bases to be revised to be consistent with the ASME OM Code, such as in paragraph d of Technical Specification Section 5.5.3, and in References 4 and 5 to Technical Specification Bases for Surveillance Requirement 3.7.1.1. In its letter dated January 26, 2010, Westinghouse provided a planned revision to the AP1000 DCD Technical Specifications and Technical Specification Bases to be consistent with the ASME OM Code. The staff considers these planned changes to the AP1000 DCD will resolve this portion of RAI-SRP3.9.6-CIB1-08.

Therefore, OI-SRP3.9.6-CIB1-05 is closed. The planned changes to the AP1000 DCD will be tracked as **Confirmatory Item CI-SRP-3.9.6-CIB1-05**.

AP1000 DCD Tier 2, Subsection 3.9.6.2.2 discusses check valve testing in a subsection titled "Check Valve Exercise Tests." The revision to the AP1000 DCD Tier 2 indicates check valves must be exercised in the open and closed directions. In RAI-SRP3.9.6-CIB1-09, the NRC staff requested Westinghouse to clarify the discussion of the AP1000 IST Program to support implementation of the AP1000 DCD provisions for check valves by a COL applicant referencing the AP1000 reactor design. In its response to this RAI, in a letter dated September 9, 2008, Westinghouse specified the acceptance criteria for assessing individual valve performance will be based on full open (full disk lift or achieving design accident flow rates) and valve closure verification using differential pressure/backflow tests. Westinghouse stated it is anticipated that Appendix II, "Check Valve Condition Monitoring Program," of the ASME OM Code will be implemented after sufficient operational data are obtained for the AP1000 check valves. The NRC staff considered the RAI response to be acceptable, but the AP1000 DCD needed to include the specified acceptance criteria for check valve testing. The staff tracked this item as OI-3.9.6-CIB1-06. In letters dated January 26 and March 5, 2010, Westinghouse provided planned changes to AP1000 DCD Tier 2, Section 3.9.6 and Table 3.9-16 to include the check valve test acceptance criteria and to identify those check valves that will need to have a mechanical exerciser installed in lieu of flow testing. The staff considers that these planned changes to the AP1000 DCD will resolve RAI-SRP3.9.6-CIB1-09. Therefore, OISRP3.9.6-CIB1-06 is closed. The planned changes to the AP1000 DCD will be tracked as **Confirmatory Item CI-SRP-3.9.6-CIB1-06**.

The subsection titled "Pressure/Vacuum Relief Devices," in AP1000 DCD, Tier 2, Subsection 3.9.6.2.2 addresses the IST Program for pressure and vacuum relief devices. In RAI-SRP3.9.6-CIB1-10, the NRC staff requested that Westinghouse provide additional information in specific areas regarding the IST Program for safety and relief valves. In response to this RAI, in a letter dated September 9, 2008, Westinghouse stated reactor coolant system pressure relief devices are discussed in AP1000 DCD Tier 2, Section 5.4.9. Pressure relief devices for other ASME Code systems are described with the applicable system in the AP1000 DCD. All safety and relief valves included in the IST Program will be tested to the rules of Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," of the ASME OM Code. ASME Code Class 1, 2, and 3 pressure relief valves are identified in AP1000 DCD Tier 2, Table 3.9-16. The NRC staff considers this clarification of the applicable provisions for safety and relief valves to be consistent with the ASME OM Code. RAI-SRP3.9.6-CIB1-10 is resolved.

AP1000 DCD Tier 2, Table 3.9-16 lists the valves in the IST Program with their valve and actuator type, safety-related missions, safety functions, ASME Class and IST Category, and IST type and frequency. In RAI-SRP3.9.6-CIB1-11, the NRC staff requested that Westinghouse update Note 31 of Table 3.9-16 that addresses operability testing of various POVs to reflect changes to the AP1000 DCD. In its response to this RAI, in a letter dated July 18, 2008, Westinghouse stated the MOV and air-operated valve (AOV) programs are expected to incorporate attributes for a successful POV periodic verification program as discussed in Regulatory Issue Summary (RIS) 2000-03, "Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves under Design Basis Conditions." Westinghouse provided a planned revision to Note 31 of Table 3.9-16 stating the applicable

valves are subject to operability testing per the NRC regulations in 10 CFR Part 50, Section 50.55a. The NRC staff considered Note 31 needed to be clarified to be consistent with the JOG MOV Periodic Verification Program, and to include the expectation indicated by Westinghouse in the RAI response that the MOV and AOV programs will incorporate attributes for a successful POV periodic verification program as discussed in RIS 2000-03. The staff tracked this item as Open Item OI-SRP3.9.6-CIB1-07. In its letter dated January 26, 2010, Westinghouse provided a planned revision to Note 31 in Table 3.9-16 that will specify valve test frequencies will be established in accordance with the results of the JOG MOV Periodic Verification Program. The planned Note 31 revision will also state the JOG approach will be applied to all actuator types and the attributes of the POV programs will include lessons learned as delineated in RIS 2000-03. The staff considers these planned changes to the AP1000 DCD will resolve RAI-SRP3.9.6-CIB1-11. Therefore, OI-SRP3.9.6-CIB1-07 is closed. The planned changes to the AP1000 DCD will be tracked as **Confirmatory Item CI-SRP-3.9.6-CIB1-07**.

The revision to the AP1000 DCD includes changes to several notes in Table 3.9-16. In RAI-SRP3.9.6-CIB1-12, the NRC staff requested that Westinghouse discuss the basis for the changes specified to Table 3.9-16. In its response to this RAI, in a letter dated September 9, 2008, Westinghouse stated Note 2 addressing valve safety functions includes such cases where normal valve operator action moves the valve to the open or closed position by de-energizing the operator electrically, by venting air, or both, then the exercise test will satisfy the fail-safe test requirements and an additional test for fail-safe testing will not be performed. Note 20 indicates the main steam isolation valves (MSIVs) and main feedwater isolation valves (MFIVs) will not be exercised during power operation to avoid a potential plant transient and reactor trip consistent with the guidance in NUREG-1482. Note 33 applies to fuel transfer tube isolation manual valve FHS-PL-V001 that will be tested consistent with 10 CFR Part 50, Section 50.55a(b)(3)(vi) at a 2 year interval. Note 38 applies to main control room emergency habitability system (VES) pressure regulating valves that are exempt from the ASME OM Code, but Westinghouse stated it would revise the note in Table 3.9-16 to clarify the testing for these valves. As a result, Revision 17 to the AP1000 DCD, Subsection 3.9.6 modifies Note 38 to state exercise stroke tests for the VES pressure regulating valves will consist of a pressure drop test across the valve using the downstream test connection to ensure adequate testing of the valves. The NRC staff finds the Westinghouse response to RAI-SRP3.9.6-CIB1-12 and Revision 17 to the AP1000 DCD, Subsection 3.9.6 adequately clarify the testing for the valves described in the applicable notes in Table 3.9-16 discussed in this RAI to be consistent with the ASME OM Code and the NRC regulations. RAI-SRP3.9.6-CIB1-12 is closed.

The revision to the AP1000 DCD Tier 2 modifies Subsection 3.9.6.2.2 in a subsection titled "Remote Valve Position Indication Inservice Tests" to state position indication testing requirements for passive valves are identified in Table 3.9-16. In RAI-SRP3.9.6-CIB1-13, the NRC staff requested that Westinghouse clarify this modification. In its response to this RAI, in a letter dated July 24, 2008 (ADAMS Accession Number ML082100164), Westinghouse stated passive valves with remote position indication will be locally observed to verify the remote position indication accurately reflects valve position. All valves requiring position indication verification will be exercised during the position indication test such that the open and closed positions can be verified. The frequency of this test will be once every 2 years. All passive valves with test requirements are included in AP1000 DCD Tier 2, Table 3.9-16. The

NRC staff considers the incorporation of passive valves with test requirements in Table 3.9-16 to be consistent with the requirements of the ASME OM Code, Subsection ISTC-3700, "Position Verification Testing." RAI-SRP3.9.6-CIB1-13 is resolved.

Subsection 3.9.6.2.2 of the AP1000 DCD Tier 2 under Manual/Power-Operated Valve Tests states the IST requirements for measuring stroke time for valves in AP1000 reactor will be completed in conjunction with a valve exercise test, and the stroke time test is not identified as a separate test. In RAI-SRP3.9.6-CIB1-14, the NRC staff requested that Westinghouse clarify the stroke time testing provisions in the AP1000 DCD. In its response to this RAI, in a letter dated July 24, 2008, Westinghouse stated each POV is stroke-time tested when the full stroke exercise test is performed. The stroke time open or closed will match the safety-related mission (i.e., transfer open or closed) as identified in AP1000 DCD Tier 2, Table 3.9-16. The NRC staff considers the IST description for stroke-time testing specified in Table 3.9-16 to be consistent with the ASME OM Code. RAI-SRP3.9.6-CIB1-14 is resolved.

Subsection 3.9.6.2.2 of the AP1000 DCD Tier 2 under Manual/Power-Operated Valve Tests states safety-related valves that fail to the safety-related actuation position to perform the safety-related missions are subject to a valve exercise inservice test. In RAI-SRP3.9.6-CIB1-15, the NRC staff requested that Westinghouse clarify the discussion of fail safe testing. In its response to this RAI, in a letter dated July 24, 2008, Westinghouse stated the exercise test will satisfy the fail safe test requirements in cases where normal valve operator action moves the valve to the open or closed position by de-energizing the operator electrically, by venting air, or both. Westinghouse indicated remote position indication is used as applicable to verify proper fail safe operation, provided the indication system for the valve is periodically verified in accordance with ASME OM Code, Subsection ISTC-3700. The valves listed in Table 3.9-16 with an Active to Failed Safety Function are designed for only one safety-related mission direction with the fail position being the transfer open or transfer close position. The NRC staff considered the reference to ASME OM Code, Subsection ISTC-3700, needed to be clarified to confirm the exercise test frequency requirements specified in the ASME OM Code for these valves will be satisfied. This item was tracked as Open Item OI-SRP3.9.6-CIB1-08. In its letter dated January 26, 2010, Westinghouse noted the Position Indication Verification Test is separate and independent of the Fail Safe Test. Westinghouse provided a planned revision to AP1000 DCD Tier 2, Table 3.9-16 to indicate a separate Fail Safe test for the applicable valves with fail safe functions. The staff considers these planned changes to the AP1000 DCD will resolve RAI-SRP3.9.6-CIB1-15. Therefore, OI-SRP3.9.6-CIB1-08 is closed. The planned changes to the AP1000 DCD will be tracked as **Confirmatory Item CI-SRP-3.9.6-CIB1-08**.

The revision to Subsection 3.9.6.2.2 of the AP1000 DCD Tier 2 under Check Valve Exercise Tests states, if exercise testing during a refueling outage is not practical, then another method is applied, such as nonintrusive diagnostic techniques or valve disassembly and inspection. In RAI-SRP3.9.6-CIB1-16, the NRC staff requested that Westinghouse clarify the revision to the AP1000 DCD for check valve exercise testing. In its response to this RAI, in a letter dated July 24, 2008, Westinghouse stated no check valves for which exercise tests are recommended have been identified, which cannot be full stroke exercised. As a result, neither nonintrusive techniques nor disassembly/inspection is required as part of the AP1000 certified design. If check valves are identified for which exercise tests are recommended but not practical due to operational issues or changes to the ASME OM Code, Westinghouse stated

it will be the responsibility of the licensee to define the types of nonintrusive diagnostic techniques to be used. To clarify this provision, Revision 17 to AP1000 DCD Tier 2, Subsection 3.9.6.2.2, specifies the check valves included in the IST Program outlined in Table 3.9-16 do not require another means as an alternate to exercise testing based on the ASME OM Code used to develop the IST plan for the AP1000 Design Certification. The NRC staff finds the Westinghouse response to RAI-SRP3.9.6-CIB1-16 and Revision 17 to the AP1000 DCD provide an acceptable clarification of the exercise testing for check valves. RAISRP3.9.6-CIB1-16 is closed.

Subsection 3.9.6.2.2 of the AP1000 DCD Tier 2 under Pressure/Vacuum Relief Devices states the frequency for this inservice test is every 5 years for ASME Class 1 and main steam safety valves, or every 10 years for ASME Classes 2 and 3 devices. The ASME OM Code also requires 20 percent of the valves from each valve group be tested within any 24 month interval for Class 1 and main steam safety valves, and within any 48 month interval for Class 2 and 3 devices. In RAI-SRP3.9.6-CIB1-18, the NRC staff requested that Westinghouse discuss the requirement to test 20 percent of each valve group within the interval required by the ASME OM Code. In response to this RAI, in a letter dated July 24, 2008, Westinghouse indicated that AP1000 DCD Tier 2, Table 3.9-16 includes the provision for 20 percent of the valves from each group to be tested. Further, Revision 17 to the AP1000 DCD Tier 2, Subsection 3.9.6.2.2, clarifies the provision that 20 percent for the valves from each group will be tested within any 24 month interval for Class 1 and main steam safety valves, and within any 48 month interval for Class 2 and 3 devices. The NRC staff finds the Westinghouse response to RAI-SRP3.9.6-CIB1-18 as incorporated into AP1000 DCD Revision 17 provides an acceptable clarification to ensure the IST activities are consistent with the ASME OM Code. RAI-SRP3.9.6-CIB1-18 is resolved.

The revision to the AP1000 DCD Tier 2 modifies Subsection 3.9.6.2.3 to state the sample disassembly examination program shall group check valves of similar design, application, and service condition, and shall require a periodic examination of one valve from each group. In RAI-SRP3.9.6-CIB1-19, the NRC staff requested that Westinghouse clarify its plans for the disassembly examination program for check valves. In its response to this RAI, in a letter dated July 24, 2008, Westinghouse stated all check valves in the AP1000 IST Program outlined in AP1000 DCD Tier 2, Table 3.9-16 are capable of being full stroke exercise tested based on the ASME OM Code (1995 Edition and 1996 Addenda) used to develop the IST plan for the AP1000 Design Certification. Westinghouse indicated it will be the responsibility of the licensee to define requirements of a disassembly and inspection program if check valves are identified for which exercise tests are recommended, but are not practical due to operational issues or changes in the ASME OM Code. The provisions in the AP1000 DCD for check valve exercise tests are consistent with ASME OM Code and, therefore, are acceptable. RAI-SRP3.9.6-CIB1-19 is resolved.

The revision to AP1000 DCD Tier 2 modifies Table 3.9-16 to identify valve type, operator, class and category for valves in the AP1000 IST Program. In RAI-SRP3.9.6-CIB1-20, the NRC staff requested that Westinghouse clarify several items in Table 3.9-16. In its response to this RAI, in a letter dated September 9, 2008, Westinghouse discussed each specific RAI item and planned changes to the AP1000 DCD. For example, Westinghouse provided a modification to Table 3.9-16 (incorporated in Revision 17 to the AP1000 DCD) that includes a provision for full stroke exercising during refueling outages for service air supply containment isolation valve

CAS-PL-V205. Westinghouse stated that the chemical volume and control system (CVS) containment isolation valves CVS-PL-V045, CVS-PL-V047, CVS-PL-V090, CVS-PL-V091, CVSPL-V092, and CVS-PL-V094 will receive only a leakage test in accordance with 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Westinghouse clarified that the air operator for reactor coolant system purification return line stop valve CVS-PL-V081 does not perform a safety function, and the valve will act as a simple check valve upon loss of power. Westinghouse provided a modification to Table 3.9-16 (incorporated in AP1000 DCD, Revision 17) that specifies full stroke exercise tests during refueling outages for demineralized water supply containment isolation check valve DWS-PL-V245 and fire water containment supply isolation check valve FPS-PL-V052. Westinghouse provided a correction to Table 3.9-16 (incorporated in AP1000 DCD, Revision 17) to reflect the 2 year test frequency and IST Category C for automatic depressurization system (ADS) discharge header vacuum relief valve RCS-PL-V010A and V010B. Westinghouse clarified that the main control room emergency habitability system pressure regulating valves VES-PL-V002A and V002B are pressure regulating valves that are not part of the ASME OM Code IST Program, and provided a modification to Table 3.9-16 (incorporated in AP1000 DCD, Revision 17) to specify these valves are part of an augmented inspection program. Westinghouse stated Note 3 in Table 3.9-16 would be revised to remove the discussion of probabilistic risk assessment for the ADS valves. Westinghouse stated the leak testing for valves CVS-PL-V001, V002, V080, V081, V082, V084, and V085 described in Note 32 is beyond the ASME OM Code IST program, and is part of an augmented testing program. Westinghouse provided a modification to Table 3.9-16 (incorporated in AP1000 DCD, Revision 17) to correct the categorization of these CVS valves from Category A to Category B or C, which do not require OM Code leak testing.

The NRC staff determined several aspects of this RAI response needed to be clarified and tracked this item as OI-SRP3.9.6-CIB1-09. Westinghouse addressed this OI in its letter dated January 26, 2010. First, Westinghouse stated CVS valves CVS-PL-V045, V047, V090, V091, V092, and V094 have a safety function to transfer closed for containment isolation and do not serve an RCS pressure boundary function. Westinghouse provided a planned revision to AP1000 DCD Tier 2, Table 3.9-16 to correct the function indication for these valves. Second, Westinghouse provided a planned revision to Note 3 in Table 3.9-16 to ensure consistency with RAI response. Third, Westinghouse clarified its response regarding the categorization of the CVS valves discussed in Note 32. The staff considers the clarifications and the planned changes to the AP1000 DCD will resolve RAI-SRP3.9.6-CIB1-20. Therefore, OI-SRP3.9.6-CIB1-09 is closed. The planned changes to the AP1000 DCD will be tracked as **Confirmatory Item CI-SRP-3.9.6-CIB1-09**.

Subsection 3.9.6.2.2 of the AP1000 DCD Tier 2 under Check Valve Low Differential Pressure Tests identifies low differential pressure testing as an inservice test is performed in addition to exercise inservice tests once each refueling cycle. In RAI-SRP3.9.6-CIB1-17, the NRC staff requested that Westinghouse clarify the discussion of low differential pressure testing. In its response to this RAI, in a letter dated July 24, 2008, Westinghouse stated the low differential pressure testing is part of an augmented test activity similar to that established for the AP600 reactor design during NRC staff review of that design certification. As a result, Revision 17 to AP1000 DCD Tier 2, Subsection 3.9.6.2.2, indicates the low differential pressure testing is not required by the ASME OM Code, but is part of an augmented inspection program. In its RAI response, Westinghouse indicated AP1000 DCD Tier 2, Table 3.9-16

will be revised to specify this test will be performed once every refueling cycle. The NRC staff finds the Westinghouse response to RAI-SRP3.9.6-CIB1-17 adequately clarifies the AP1000 test activities to be consistent with the AP600 certified design. However, the planned changes to Table 3.9-16 for the applicable check valves do not appear to be included in Revision 17 to the AP1000 DCD. This item will be tracked as **Confirmatory Item CI-SRP3.9.6-CIB1-10**.

The revision to AP1000 DCD Tier 2 includes a new Subsection 3.9.3.4.4, "Inspection, Testing, Repair and/or Replacement of Snubbers," which specifies a program for inservice examination and testing of dynamic supports (snubbers) to be used in the AP1000 reactor will be prepared in accordance with the requirements of ASME OM Code, Subsection ISTD. AP1000 DCD Tier 2, Subsection 3.9.3.4.4 indicates details of the snubber inservice examination and testing program, including test schedules and frequencies, will be reported in the inservice inspection and testing plan included in the IST Program required by AP1000 DCD Tier 2, Subsection 3.9.8.3, "Snubber Operability Testing." AP1000 DCD Tier 2, Subsection 3.9.8.3 states a COL applicant referencing the AP1000 design will develop a program to verify operability of essential snubbers. The NRC staff finds the provision in the AP1000 DCD for application of the ASME OM Code, Subsection ISTD, in the examination and testing of dynamic supports to be acceptable for the AP1000 Design Certification. The COL applicant will be responsible for satisfying the COL Information Item in AP1000 DCD Tier 2, Subsection 3.9.8.3.

The staff reviewed the revisions to the AP1000 DCD with respect to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to be used at an AP1000 nuclear power plant. The staff finds the changes are generic and are expected to be applicable to all COL applications referencing the AP1000 certified design.

3.9.6.2 Conclusion

The NRC staff concludes the revision to the AP1000 DCD continues to support the design aspects for the functional design, qualification, and IST programs for safety-related valves and dynamic restraints in the applicable NRC regulations for the AP1000 Design Certification. The revision to the AP1000 certified design provides sufficient information to satisfy 10 CFR Parts 50 and 52 for the design aspects of the functional design, qualification, and IST programs for safety-related valves and dynamic restraints to be used in the AP1000 reactor, pending resolution of the identified confirmatory items. The NRC staff will review the operational program aspects regarding the functional design, qualification, and IST programs for safetyrelated valves and dynamic restraints in a COL application referencing the AP1000 certified design as part of the COL application review process.

3.9.7 Integrated Head Package

The integrated head package (IHP) provides the ability to rapidly disconnect cables including the CRDM power cables, digital rod position indication cables, and in-core instrument cables from the IHP components. The rapid disconnection of these cables provides the ability to move the IHP components as an assembly to permit the expedited lifting and removal of the reactor vessel head. In addition, the IHP provides support for the vessel head stud tensioner/detensioner during refueling. The IHP includes a lifting rig, seismic restraints for CRDMs, and support for the following IHP components: reactor head vent piping, cable bridge,

power cables, cables and guide tubes for in-core instrumentation, cable supports, and shroud assembly.

By letter dated November 14, 2006, Westinghouse submitted TR-61, "AP1000 Integrated Head Package," APP-GW-GLN-014 (ADAMS Accession Number ML063210447). The purpose of TR-61 was to address changes in the IHP described in Revision 15 to the AP1000 DCD as reviewed by the staff in NUREG-1793.

Following a preliminary review, the staff requested additional information in a March 29, 2007 letter (ADAMS Accession Number ML070850160), via questions RAI-TR61-01 through RAI-TR61-04. By letter dated April 13, 2007 (ADAMS Accession Number ML071070483), Westinghouse provided responses to the staff's questions. It should be noted much of the staff's focus in the review of TR-61 was associated with the change in the IHP design related to the removal of the CRDM cooling fans from the IHP to a separate structure and the resulting questions related to the adequacy of CRDM cooling.

Westinghouse subsequently submitted Revisions 16 and 17 to the DCD. In Subsection 3.9.7 of Revision 17 to the DCD, Westinghouse, again, proposes to attach the CRDM cooling fans to the IHP. In addition, the following changes are proposed:

- In the first paragraph of Subsection 3.9.7, the cable bridge is included in the IHP description but the guide tubes for in-core instrumentation are excluded.

- In Subsection 3.9.7.1, the shroud and CRDM seismic support plate, are no longer in the list of components which are required to provide seismic restraint for the CRDM and the valves and piping of the reactor head vent. The CRDM and the valves and piping of the reactor head vent still require seismic restraints. These components are AP1000 equipment Class C, seismic Category I and are designed in accordance with the ASME Code, Section III, Subsection NF requirements.

- The instrumentation guide tubes and the instrumentation support structure are excluded from those components function as part of the lifting rig and are required to be capable of lifting and carrying the total assembled load of the IHP.

- The components of the in-core instrumentations system (IIS) that interface with the IHP are the QuickLoc stalk assembly and the IIS cables and connectors. These have been excluded from the IHP description.

- The shroud assembly is required to provide radiation shielding of the CRDMs but the conduit for in-core instrumentation when the instrumentation is withdrawn into the conduit is not required to provide shielding. The radiation level at the exterior surface of the shroud during refueling with the in-core instrument thimble withdrawn is excluded from the discussion in the radiation levels discussed in Section 12.2.

- The description of the IHP in Subsection 3.9.7.2 excludes the In-core Instrumentation support structure.

- The description of the lifting system is modified. The lifting system attaches to the CRDM seismic support structure. The lift lugs transfer the head load during a head lift from the head attachment lugs; however, the attachment is no longer through the CRDM seismic support structure to the lift rig.

- In the description of the mechanism seismic support structure has been modified to reflect minor, proposed changes in the support structure.

- The description of the In-core Instrumentation-support structure (IISS), has been changed to discuss the In-core Instrumentation. The following statements related to the support structure have been deleted:

  - The in-core instrumentation support structure is used during refueling operations. This support structure is used for withdrawing the in-core instrumentation thimble assemblies into the integrated head package. It protects and supports the thimble assemblies when they are in the fully withdrawn position.

  - Also, the in-core instrumentation support structure includes a platform which provides access to the in-core instrumentation during maintenance and refueling and to attach the lifting system to the crane hook.

3.9.7.1 Evaluation

The staff reviewed the proposed changes related to Subsection 3.9.7 of AP1000 DCD Revision 17, including TR-61. The AP1000 IHP continues to meet all applicable acceptance criteria and requirements, as discussed below. The components of the IHP, which provide seismic support including the CRDM seismic support and the shroud, are designed using the ASME Code, Section III, Subsection NF which satisfies the limit on deflection of the top of the CRDM rod travel housing. The components of the IHP included in the load path of the lifting rig are designed to satisfy the requirements for lifting of heavy loads in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980. The criteria of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or More for Nuclear Materials," are used to evaluate the loads and stresses during a lift.

Those cables and connectors for the IIS are required to meet Class 1E requirements are evaluated for environmental conditions including normal operation and postulated accident conditions.

Components required to provide seismic restraint for the CRDMs and the valves and piping of the reactor head vent are AP1000 equipment Class C, seismic Category I and are designed in accordance with the ASME Code, Section III,Subsection NF requirements.

The loads and loading combinations due to seismic loads for these components are developed using the appropriate seismic spectra.

The structural design of the IHP is based on a design temperature consistent with the heat loads from the vessel head, the CRDMs, and electrical power cables. The design also considers changes in temperature resulting from plant design transients and loss of power to the cooling fans.

Components required to provide cooling to the CRDMs are non-safety-related AP1000 equipment Class E. Section 4.6 of the DCD Revision 17, offers a discussion of the effect of failure of cooling of the CRDMs.

Those components that function as part of the lifting rig are required to be capable of lifting and carrying the total assembled load of the IHP which includes the vessel head, CRDMs, CRDM seismic supports, shroud, cooling ducts, and insulation. The lifting rig components are required to meet the guidance for special lifting rigs, in NUREG-0612. The lifting rig components are non-safety-related, AP1000 equipment Class E.

The electrical cables and connectors, within the IHP, for the IIS are AP1000 equipment Class C, Class 1E. The other cables within the IHP, including power cables and cables for the digital rod position indicator system, are not Class 1E. The cable support provides seismic support and maintains separation for instrumentation and power cables.

3.9.7.2 Conclusion

The components of the IHP, which provide seismic support including the CRDM seismic support and the shroud, are designed using the ASME Code, Section III, Subsection NF. The IHP satisfies the limit on deflection of the top of the CRDM rod travel housing. The components of the IHP included in the load path of the lifting rig are designed to satisfy the requirements for lifting of heavy loads in NUREG-0612. The criteria of ANSI N14.6, are used to evaluate the loads and stresses during lifting.

Those cables and connectors for the IIS are required to meet Class 1E requirements are evaluated for environmental conditions including normal operation and postulated accident conditions. Accordingly, the staff concludes the AP1000 IHP design meets the requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, and 30; and 10CFR Part 50, Appendix S; therefore, the proposed changes to Subsection 3.9.7 of AP1000 DCD Revision 17 are acceptable.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

In Revision 17 of the DCD, Section 3.10, the applicant proposed some editorial and technical changes and clarifications. A summary of the major changes is described below.

One of the significant changes from DCD Revision 15 to DCD Revision 17 is that Westinghouse decided not to use Experience-Based Qualification Method for seismic qualification of AP1000 mechanical and electrical equipment. Therefore, all statements related to the experience-based qualification have been deleted or revised. For example, Section 3.10.6 and Item E.7 of Attachment E of Appendix D have been deleted.

In the introductory statements for Section 3.10 of AP1000 DCD Revision 17, a new paragraph was added to address the CSDRS exceedance in the high frequency spectrum region at some Central and Eastern United States rock sites. A new Reference 3 was added to DCD Revision 17 and this new Reference 3 (Not "Reference 5" as indicated in the new paragraph) is related to the "AP1000 Design Control Document High Frequency Seismic Tier 1 Changes." The Tier 2 material related to the high frequency seismic input is provided in AP1000 DCD Revision 17, Appendix 3I.

Appendix 3I of AP1000 DCD addresses the effect of HRHF seismic input. The AP1000 HRHF evaluation study is reported in TR-115, "Effects of High Frequency Seismic Contention on SSCs," (APP-GW-GLR-115) and TR-115 is referenced in AP1000 DCD Revision 17. In the course of reviewing TR-115, staff generated a list of RAIs which is applicable to DCD Appendix 3I of AP1000 DCD Revision16 and Revision 17.

3.10.1 Evaluation

The staff reviewed the major changes to Section 3.10 of the AP1000 DCD Revision 17 in accordance with the guidance in (1) the SRP Section 3.10, "Seismic and Dynamic Qualification of Mechanical Electrical Equipment," (2) COL/DC-ISG-1, "Interim staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications," May 19, 2008, and (3) SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Design," April 2, 1993, including SRM93-087 issued on July 21, 1993. The regulatory basis for Section 3.10 of the AP1000 DCD is documented in NUREG-1793

The changes in Appendix 3I related to Section 3.10 are mainly provided in Subsection 3I.6.4. The changes involve editorial clarifications and technical revisions. The results of the staff's review of the list of RAI responses are described below.

In RAI-SRP3.10-EMB-01, the staff requested the applicant (Westinghouse) to describe the screening process for potential high frequency sensitive mechanical and electrical equipment and components, and to provide a list of equipment including the justification for screening in or screening out. The detailed response to this RAI was initially submitted under Westinghouse letter DCP/NRC2144, dated May 28, 2008 (ADAMS Accession Number ML081540037), and later, a revised response was provided under Westinghouse letter DCP/NRC2235, dated August 21, 2008 (ADAMS Accession Number ML082390116). Westinghouse stated the AP1000 screening process for potential high frequency sensitive equipment is consistent with the US NRC requirements in Section 4.0 (Identification and Evaluation of HF Sensitive Mechanical and Electrical Equipment/Components) of Interim staff Guidance, COL/DC-ISG-1, and guidelines identified in the EPRI White Paper, "Considerations for NPP Equipment and Structures Subjected to Response Levels Caused by High Frequency Ground Motions," transmitted to the NRC on March 16, 2007.

The goal of the AP1000 HRHF screening program is to identify those safety-related equipment and components are potentially HRHF-sensitive and show them to be acceptable for their specific application (screened-out). The AP1000 HRHF screening program is a two-step process; the first step is a HRHF susceptibility review to identify potential high frequency sensitive safety-related equipment. The second step is the screened-out equipment process to

demonstrate its acceptability for the HRHF seismic excitation. Evaluation of screened-in equipment as defined in COL/DC-ISG-1 is not performed because all safety-related equipment is screened-in will be eliminated or shown to be acceptable through a design change process. Additional information is provided in Appendix 3I.6.4 of AP1000 DCD, Revision 17.

The staff reviewed Westinghouse's response related to the criteria and procedure for the AP1000 HRHF screening program as described above, and finds the response to be acceptable. The staff considers RAI-SRP3.10-EMB-01 to be closed.

In RAI-SRP3.10-EMB-02, the staff requested the applicant to explain, with respect to TR-115 Section 6.4.5, "Screening Process," its justification for using 50 Hz as the cut-off natural frequency for the Group No. 1 rugged equipment in the screening process, and to explain whether the electrical/electronic equipment/devices with natural frequencies greater than 50 Hz are considered as rugged equipment. The staff also requested the applicant to provide justification for not requiring additional evaluation for high frequency seismic inputs for equipment.

In Section 6.4.5 of TR-115 for the Screening Process, Westinghouse concluded safety-related equipment may be screened and grouped as follows: Group No.1 – Rugged equipment with dominant natural frequencies above 50 Hz; Group No. 2 – Cabinets and other equipment which exhibit dominant natural frequencies below HRHF exceedance range; and Group No. 3 – safety-related equipment which exhibit dominant natural frequencies in HRHF exceedance range. For Group No. 1 and Group No. 2 equipment, no additional evaluation for high frequency seismic input is necessary. For Group No. 3 equipment, the equipment will be subjected to supplemental high frequency seismic evaluation to verify acceptability.

The response to RAI-SRP3.10-EMB-02 was submitted under Westinghouse letter DCP/NRC 2144, dated May 28, 2008 (ADAMS Accession Number ML081540037). Westinghouse stated that, for AP1000, the frequency range of interest in the screening process is 25 Hz to 50 Hz. This range coincides with the peak region of the HRHF ground motion. Since the AP1000 plant building structure's dominant natural frequencies are considerably lower than 50 Hz, the horizontal and vertical ground motion response spectra (GMRS) above 50Hz will not be amplified significantly and their response will dissipate quickly as it travels through the building structure. The worst case seismic loading will occur when the fundamental frequencies of the potential HRHF-sensitive equipment coincide with the peak of the response spectra. In addition, Westinghouse noted from review of AP1000 HRHF in-structure response spectra (ISRS) generated from the HRHF ground motions above 50 Hz, the zero period acceleration (ZPA) regions of the response spectra are being approached. Westinghouse further stated equipment designs with dominant natural frequencies above 50 Hz are inherently rugged. The highly unlikely case of HRHF-sensitive equipment with a natural frequency of 55 Hz, for example, is a special class and would require combining screening process Groups Nos. 1 and 3. For this condition, the Group No. 3 process would govern and the equipment would be subjected to a supplemental HRHF seismic evaluation/screening test.

The staff concludes that, in general, 50 Hz is adequate to be used as the cut-off frequency for rugged equipment in screening process if the ZPA of the HRHF ISRS approaches 50 Hz. The staff considers RAI-SRP3.10-EMB-02 to be closed.

In RAI-SRP3.10-EMB-03, the staff requested the applicant to provide justifications for not performing additional low level testing (5 OBEs) for equipment identified as potentially sensitive to high frequency motion is located in an area with potential for high frequency seismic input motions. OBE testing requirements of IEEE Std. 344-1987 and SRP Section 3.10 must be satisfied. The NRC's policy and staff's technical positions related to OBE issues are clearly delineated in SECY-93-087. The detailed response to RAI-SRP3.10-EMB-03 was initially submitted under Westinghouse letter DCP/NRC2144, dated May 28, 2008 (ADAMS Accession Number ML081540037), and later, a revised response was provided under Westinghouse letter DCP/NRC2235, dated August 21, 2008 (ADAMS Accession Number ML082390116). In the May 28, 2008, response, Westinghouse stated the HRHF screening test is not considered to be a qualification test. The HRHF screening test is intended as a supplemental test to the required seismic qualification performed in accordance with IEEE 344. As a result of further discussion with the NRC staff, Westinghouse submitted its revised response on August 21, 2008 (ADAMS Accession Number ML082390116). Westinghouse stated its HRHF screening test will be in compliance with the seismic test input requirements in IEEE Std 344-1987 and Interim Staff Guidance defined in COL/DC-ISG-1. The five OBE (one-half SSE) and a minimum of one SSE AP1000 CSD ISRS test runs preceding the HRHF screening test are performed in compliance with IEEE Std 344-1987. All of these test runs can be used to address seismic aging (fatigue) of the safety-related equipment in the high frequency exceedance region. Each test run will produce a number of peak stress magnitudes, which will have fatigue damage potential. OBE testing in the HF exceedance region was not significant because the cyclic fatigue of equipment (ten peak stress cycles per event) for equipment is more damaging in the frequency range below the HF exceedance region. The acceleration response in the HF exceedance region will produce very small displacements and lower number of high-stress cycles resulting in the overall equipment accumulative fatigue being less than or equal to that experienced during qualification testing.

Westinghouse's response to this RAI was partially acceptable. Westinghouse did not demonstrate OBE testing requirements of IEEE Std 344-1987 and SRP 3.10 (including SECY-93-087) were satisfied. Therefore, the staff followed up with RAI-SRP3.10-EMB-10 to continue resolution of the staff concerns. The staff's evaluation of Westinghouse response to RAI-SRP3.10-EMB-10 is described later in this SER. The staff considered RAI-SRP3.10-EMB-03 to be closed.

In RAI-SRP3.10-EMB/EEB-04, the staff requested the applicant to confirm battery chargers and inverters with digital components are included in the high frequency seismic screening process. The detailed response to this RAI was initially submitted under Westinghouse letter DCP/NRC2144, dated May 28, 2008, and later, a revised response was provided under Westinghouse letter DCP/NRC2235, dated August 21, 2008. Westinghouse stated electronic components such as those found in battery chargers, inverters, and solid state and microprocessor-based components are listed in Table 6.4.5-1, "Potential Sensitive Equipment List," of TR-115. Westinghouse further stated Table 3.11-1 of AP1000 DCD Revision 16 was reviewed to verify all potential high frequency (HF) sensitive AP1000 safety-related equipment were included in APP-GW-GLN-144 (TR-144) Table A-1, "Potential High Frequency Sensitive AP1000 Safety-Related Equipment." As a result of its review, Westinghouse identified additional equipment may be potentially HF-sensitive. Table 3I.6-2 of AP1000 DCD Revision 17 and Table A-1 of TR-144 have been updated to include the following additional equipment types: batteries, neutron detectors, radiation monitors and hot leg sample isolation limit

switches. The remaining AP1000 safety-related equipment not high frequency sensitive is defined in APP-GW-GLN-144 Table A-2, "List of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive." Table 3I.6-3 of AP1000 DCD Revision 17 and Table A-3 of TR-144 include justifications for classifying the equipment as not HF-sensitive.

The staff has verified those electronic components in question are included in the tables mentioned above. The staff considers RAI-SRP3.10-EMB/EEB-04 closed.

In RAI-SRP3.10-EMB-05, the staff requested the applicant to provide justification for the conclusions addressing the use of existing test data in Section 6.4.7 (Summary and Conclusions) of TR-115. The detailed response to this RAI was initially submitted under Westinghouse letter DCP/NRC2144, dated May 28, 2008, and later a revised response was provided under Westinghouse letter DCP/NRC2235, dated August 21, 2008. Westinghouse stated the conclusions reached were based on the information presented in TR-115, Section 6.4.4 (Review of Existing Seismic Test Data). The test data in TR-115 represents existing Westinghouse seismic test data reviewed as part of the study to confirm seismic qualification to the AP1000 certified design ISRS envelops the HRHF seismic inputs for most applications. Westinghouse further stated Power Spectral Density (PSD) and other acceptable evaluation methods as defined in IEEE Std 344-1987 are ways of determining energy content within a seismic test run. When available, PSD plots were used to evaluate seismic test data reported in Section 6.4.4 of TR-115. For the test data reported, energy content in the 25 Hz to 50 Hz frequency range was demonstrated by meeting at least one of the following criteria:

1. Test reports stated the seismic time history inputs were developed with content in the frequency range up to 50 Hz as a minimum.
2. The test response spectra (TRS) were shown to be amplified in the 25 Hz to 50 Hz frequency and were not caused by impact or test unit rattling.
3. PSD plots indicate energy content in the high frequency region.

Figures 1 through 6 of the Westinghouse response (DCP/NRC2235) provide examples of test data which demonstrate frequency content in the 25 Hz to 50 Hz range.

The staff has examined Figures 1 through 6 and concluded that, for the existing test data reported, energy content in the 25 Hz to 50 Hz frequency range was demonstrated by meeting at least one of the criteria described above. Therefore, the staff considers RAI-SRP3.10-EMB-05 to be closed.

In RAI-SRP3.10-EMB-06, the staff requested the applicant to provide detailed evaluation comparisons for the reactor vessel internals response to the HRHF and CSDRS seismic input motions, and, also, the seismic anchor motion effects of the high frequency input motion. The detailed response to this RAI was submitted under Westinghouse letter DCP/NRC2152, dated June 6, 2008 (ADAMS Accession Number ML081620074). Westinghouse provided a comparison between the CSDRS results and HRHF results for various support (interface) loads within the reactor internals system model. The comparison indicates these support loads are reduced for HRHF evaluation when compared to the CSDRS analysis. The comparison also indicates CSDRS would control the cyclic loading demand. Westinghouse further stated the

seismic anchor motion effects are included in the high frequency input motion study, and therefore included in the evaluation.

The staff finds Westinghouse's response to be adequate in resolving its concerns relating to the comparison of the pertinent stress analysis results for the reactor internals system under the CSDRS and HRHF seismic input excitations. Westinghouse has also included the cyclic loading and seismic anchor motion effects in the HRHF evaluations. The staff considers RAI-SRP3.10-EMB-06 to be closed.

In RAI-SRP3.10-EMB-07, the staff requested the applicant to provide justification for concluding the reactor internals are representative of the primary mechanical components such that all others can be screened out, and also provided quantitative evaluation result for mechanical component other than reactor vessel internals to substantiate the justification. The detailed response to this RAI was submitted under Westinghouse letter DCP/NRC2152, dated June 6, 2008. Westinghouse stated the mechanical components listed in Table 3.2-3 of the AP1000 DCD, Tier 2 must be designed for the SSE are those classified as Seismic Category, I and II. Among those equipment and components, Westinghouse stated many mechanical components and equipment that are safety-related are not high frequency sensitive as is some electrical equipment. Therefore, it is only necessary to evaluate a representative sample of mechanical components and equipment. Westinghouse stated the reactor vessel is representative of a mechanical component with complex internals that was evaluated as part of the HRHF evaluation. The seismic response of this component is considered representative of other mechanical components. The reactor internals were chosen for evaluation because this is an important component related to safety, and the reactor internals are representative of other component internals. It is, therefore, not necessary to perform further analysis of other mechanical components and equipment for the HRHF earthquake excitations.

The staff concludes reactor internals are relatively complex and contain broader natural frequencies than other mechanical components. The staff considers Westinghouse's response adequate in justifying that reactor internals can be considered as representative of ASME safety-related mechanical components and the equipment for high frequency evaluation. Therefore, the staff considers RAI-SRP3.10-EMB-07 to be closed.

In RAI-SRP3.10-EMB-08, the staff requested the applicant to justify the use of the required input motion (RIM) curve of IEEE 382-1996 for qualification of line-mounted equipment (e.g., valves) for HRHF response spectra with exceedance, or to provide methodologies would be acceptable for the case of HRHF excitation. The detailed response to this RAI was initially submitted under Westinghouse letter DCP/NRC2144, dated May 28, 2008, and later Revision 1 response was provided under Westinghouse letter DCP/NRC2235, dated August 21, 2008. Revision 2 of the Westinghouse response was submitted under DCP/NRC2503, dated May 27, 2009 (ADAMS Accession Number ML091520090). Westinghouse stated it is performing seismic qualification of safety-related SSCs based on AP1000 CSDRS. The HRHF screening is a functional verification test in compliance with Interim Staff Guidance defined in COL/DC-ISG-1 to verify potential high frequency sensitive safety-related equipment will perform its function as required under Hard Rock High Frequency seismic demand response spectra. The HF screening is a supplemental evaluation to the required seismic qualification methods performed in accordance with IEEE Std. 344-1987 for those plants that have potentially high frequency sensitive equipment and components with high frequency exceedance of their CSDRS.

Westinghouse stated in those instances where the seismic qualification of line-mounted equipment (e.g., valves and their appurtenances) are potential HRHF-sensitive components, seismic testing performed in compliance with Figure 6 (RIM curve) of IEEE Std. 382-1996 will be extended out for one additional octave to 64 Hz.

AP1000 DCD Tier 2, Section 3.7.3.5.1 defines rigid components such as rigid valves as the following: "A rigid component (fundamental frequency >33 hertz), whose support can be represented by a flexible spring, can be modeled as a single degree of freedom model in the direction of excitation (horizontal or vertical directions)." When dealing with HRHF sites we should refrain from using the wording rigid equipment or rigid components because it can differ between the AP1000 CSRDS and HRHF sites. Seismic qualification of safety-related equipment by analysis will be addressed over the range of interest up to the cutoff frequency of the AP1000 certified design ISRS. In most instances a dynamic analysis or a static coefficient analysis using the peak of the applicable response spectra at the mounting location of the equipment will be used.

Westinghouse further noted in its Revision 2 Response, dated May 27, 2009, AP1000 DCD Tier 2, Appendix 3I, Table 3I.6-3 contains a list of AP1000 safety-related equipment and mechanical equipment not high frequency sensitive. Notes 1 and 2 of the table identify the requirement for performing seismic RIM testing of line-mounted equipment out to 64 Hz.

Based on the review of Westinghouse's documents, DPC/NRC2144, DCP/NRC2235, and DCP/NRC2503, the staff determined Westinghouse has adequately addressed the questions raised in this RAI. The staff has also verified the conclusion of Westinghouse's response to this RAI has been documented in Notes 1 and 2 of Table 3I.6-3 in Tier 2 document Appendix 3I of AP1000 DCD Revision 17. Therefore, the staff considers RAI-SRP3.10-EMB-08 to be closed.

In RAI-SRP3.10-EMB-09, staff requested the applicant to discuss the basis for deleting references to dampers in Section 3.10. In several locations in Section 3.10 of AP1000 DCD and Revision 17, Westinghouse has replaced the reference to safety-related dampers with a reference to safety-related valves; Subsection 3.10.2.2 is an example. Westinghouse's response to this RAI was submitted under Westinghouse letter DCP/NRC2144, dated May 28, 2008. Westinghouse stated for the AP1000 design, there are no safety-related dampers. The term "dampers" was used in error. Changes were made in Section 3.10 of AP1000 DCD and Revision 17 to correctly identify the subject equipment as safety-related valves. The staff considers Westinghouse's response to be acceptable. Therefore, RAI-SRP3.10-EMB-09 is closed.

In the revised response to NRC RAI-SRP3.10-EMB-03 dated August 21, 2008 (ADAMS Accession Number ML082390116), Westinghouse indicated the five OBE (one-half SSE) and a minimum of one SSE AP1000 CSD ISRS test runs preceding the HRHF screening test were performed in compliance with IEEE Std 344-1987. The staff understands the same specimen is used for all these test runs. Westinghouse also indicated all of the CSDRS test runs can be used to address seismic aging of the equipment in the high frequency exceedance region.

In RAI-SRP3.10-EMB-10, as a follow-up to the August 2008 response to RAI-SRP3.10-EMB-03, the staff requested Westinghouse to provide justifications including the results from calculations

that show seismic qualification of electrical/electronic equipment by tests for AP1000 CSDRS design spectra can be considered as equivalent to or more than 5 OBE peak stress cycles for HRHF spectra. This should be done using bounding AP1000 ISRS generated from CSDRS and bounding ISRS generated from HRHF Spectra and following the guidelines as delineated in Annex D of IEEE 344-1987. The staff also requested Westinghouse to document the conclusion of the comparison result of CSD ISRS and HRHF ISRS peak stress cycles in DCD Section 3I.6.4. Westinghouse's response to this RAI was submitted under Westinghouse letter DCP/NRC2280, dated October 17, 2008 (ADAMS Accession Number ML082960402) and letter DCP/NRC2396, dated March 5, 2009 (ADAMS Accession Number ML090690534). In its response, Westinghouse stated the AP1000 safety-related equipment will be seismically-qualified to the AP1000 CSD ISRS associated with the mounting location of the equipment as a minimum. Seismic qualification testing will consist of five AP1000 ISRS operating basis earthquakes (OBEs) followed by one Safe Shutdown Earthquake (SSE) as a minimum. The OBE level will be at least one-half the SSE level. The OBE testing is used to account for vibration aging and address low-cycle fatigue of equipment prior to SSE testing. Westinghouse stated cyclic fatiguing of equipment for the Hard Rock High Frequency exceedance area can be adequately addressed by performing five AP1000 ISRS OBE (one-half the SSE) and a minimum of one SSE seismic test runs in compliance with IEEE Standard 344-1987 prior to performing the supplemental HRHF screening test.

Westinghouse has performed an evaluation to demonstrate OBE testing in the high frequency exceedance range is adequately addressed by AP1000 CSD ISRS seismic qualification testing (5 OBE and 1 SSE). The evaluation compared the peak stress cycles resulting from five one-half SSE events from AP1000 HRHF ISRS to the peak stress cycles resulting from five one-half SSE events and one full SSE event from AP1000 CSD ISRS using the guidelines defined in Annex D of IEEE Std 344-1987. The Westinghouse evaluation of AP1000 CSD ISRS peak stress cycles to the AP1000 HRHF ISRS peak stress cycles is documented in Westinghouse Calculation CN-EQT-08-35 / APP-GW-S2C-002. The evaluation of AP1000 CSD ISRS peak stress cycles to the AP1000 HRHF ISRS peak stress cycles was performed for two AP1000 plant elevations; the AP1000 NI Auxiliary and Shield Building (ASB) at or below 135 feet elevation and the AP1000 Containment Internal Structure (CIS) at or below 134.25 feet elevation.

The peak stress cycles in each direction were determined based on the zero period acceleration (ZPA) of the 1/2 SSE HRHF ISRS and the 1/2 SSE and SSE CSD ISRS acceleration time histories normalized to the same ZPA value to demonstrate equivalency of results. Results of the cycle counting in compliance with guidelines defined in Annex D of IEEE Std 344-1987 are summarized in Table 1 of the Westinghouse letter, DPC/NRC2396.

Westinghouse concluded that the completed evaluation has demonstrated the peak stress cycles resulting from five one-half SSE events using the AP1000 HRHF ISRS are equivalent to or enveloped by the peak stress cycles resulting from five one-half SSE events and one full SSE event using the AP1000 CSD ISRS.

The staff has reviewed Westinghouse's responses as stated above. The staff concludes Westinghouse has adequately demonstrated by calculations that the peak stress cycles resulting from five one-half SSE events using the AP1000 HRHF ISRS are equivalent to or enveloped by the peak stress cycles resulting from five one-half SSE events and one full SSE

event using the AP1000 CSD ISRS. Westinghouse's response (DPC/NRC2396) also included the proposed revision to the DCD. The staff finds the proposed revision acceptable. However, the proposed DCD revision has not been incorporated into AP1000 DCD Revision 17 and it should be incorporated into the future DCD revision. Therefore, the staff created **CI-SRP3.10-EMB-10** to track the completion of this confirmatory Item.

In RAI-SRP3.10-EMB-11, the staff pointed out that Westinghouse's evaluations from the envelope response spectra (ERS) for the Central and Eastern U. S. rock sites (HRHF foundation response spectra exceeds CSDRS for frequencies above about 15 Hz) presented in TR-115, Revision 2, indicated some resulting floor response spectra (FRS) are higher than those presented in Revision 0 and Revision 1. Some resulting HRHF FRS exceed the CSDRS FRS significantly in the frequency range from 6 Hz to 60 Hz. Westinghouse is requested to demonstrate all AP1000 safety-related mechanical and electrical equipment (not limited to HF-sensitive equipment only) are seismically qualified to the required response spectra (RRS) at the equipment/device locations to meet the requirements of GDC 2 as a result of TR-115, Revision 2.

The detailed response to this RAI was initially submitted under Westinghouse letter DCP_NRC_003004, dated August 6, 2010, and later, Revision 1 response was provided under Westinghouse letter DCP_NRC_003019, dated August 23, 2010. Westinghouse stated the equipment seismic qualification information provided in Revision 2 of TR-115 was developed using the latest revision of the CSDRS and HRHF. The equipment will be qualified to the latest AP1000 CSDRS applicable to its mounting location. Potential HRHF sensitive equipment will be subjected to a high frequency screening test consistent with the guidance in Interim Staff Guidance (ISG) 01 after completion of CSDRS testing. Appendix 3I of the AP1000 DCD provides two tables that separate the AP1000 safety related electrical and mechanical equipment into two categories: (1) Table 3I.6-2 includes equipment that is potentially high frequency sensitive and (2) Table 3I.6-3 contains equipment not high frequency sensitive. Westinghouse stated seismic qualification for both categories is performed using the current revision of the CSDRS ISRS associated with the mounting locations of equipment based on the guidance of IEEE Std 344-1987. After completing CSDRS ISRS seismic qualification testing, the potential high frequency sensitive equipment will be subjected to a HRHF screening test to the HRHF ISRS associated with the mounting location of the equipment as a minimum. To demonstrate acceptability for both CSDRS and HRHF testing, the test response spectra must envelop the CSDRS ISRS with margin over the frequency range of interest in compliance with IEEE Std 344-1987. If the HRHF screening test cannot demonstrate the equipment to be acceptable, then the safety related equipment will be removed or modified and additional testing or justification will be required.

Westinghouse noted that, at locations where HRHF response spectra show exceedance of the CSDRS for category (2) equipment (not high frequency sensitive) in Table 3I.6-3 of AP1000 DCD, further evaluations would be performed to verify the existing qualification is adequate. Westinghouse further stated that, in the event that the CSDRS and/or HRHF response spectra would be revised after the qualification program has been completed, a reconciliation effort would be performed to verify the CSDRS and HRHF testing is still valid. The reconciliation effort may result in re-qualification activities and qualification documentation revisions.

The seismic qualification testing for both CSDRS and HRHF spectra for AP1000 safety-related equipment will be documented in equipment qualification document packages as described in Appendix 3D of the AP1000 DCD. These qualification document packages will be used to satisfy the equipment's Seismic Category I ITAAC described in Tier 1 of the AP1000 DCD. Westinghouse's response to RAI-SRP3.10-EMB-11 (DCP_NRC_003004 and DCP_NRC_003019) also included the proposed revision to the DCD (Section 3I.6.4, "Electrical and Electro-mechanical Equipment").

Based on the aforementioned response, the staff finds Westinghouse's response to RAI-SRP3.10-EMB-11 and the proposed revision to AP1000 DCD are acceptable. Therefore, the staff created **CI-SRP3.10-EMB-11** to track the completion of this confirmatory item.

3.10.2 Conclusion

The staff reviewed the proposed changes related to Section 3.10 of AP1000 DCD Revision 17, including Appendix 3I to the DCD. On the basis that the AP1000 mechanical and electrical equipment continue to meet all applicable acceptance criteria and procedures for seismic qualification of mechanical electrical equipment in accordance with the guidance of SRP Section 3.10, RG 1.100, SECY-93-087, and COL/DC-ISG-1, the staff finds the changes to Section 3.10 of AP1000 DCD Revision 17 are acceptable subject to **Confirmatory Items CI-SRP3.10-EMB-10 and CI-SRP3.10-EMB-11**. The staff finds the AP1000 design provides adequate assurance that AP1000 Seismic Category I equipment will function properly under the effects of earthquake motions, and the acceptance criteria for the AP1000 design meet the requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, and 30; and 10 CFR Part 50, Appendix S.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

In Revision 17 to the AP1000 DCD Tier 2, Westinghouse modified Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." The objective of environmental qualification (EQ) is to reduce the potential for common failure due to specified environmental conditions and seismic events, and to demonstrate the equipment within the scope of the EQ Program is capable of performing its intended design safety function under all conditions including environmental stresses resulting from design bases events.

3.11.1 Evaluation

In Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," of NUREG-1793, the NRC staff described its review of the description of the EQ Program for the AP1000 design. The regulatory basis for the NRC review of the design certification information is documented in NUREG-1793. The regulatory basis for the proposed changes to the AP1000 DCD is the same as specified in NUREG-1793.

In NUREG-1793, the staff concluded the program described for environmentally qualifying electrical equipment important to safety and safety-related mechanical equipment in support of the AP1000 Design Certification complied with the requirements for 10 CFR 50.49 and other relevant requirements and criteria.

Since the issuance of NUREG-1793, the NRC has determined a COL applicant referencing the AP1000 design needs to fully describe EQ and other operational programs as defined in Commission Paper SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria." RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," provides guidance for a COL applicant in preparing and submitting its COL application in accordance with the NRC regulations. For example, Section C.IV.4 in RG 1.206 discusses the requirement in 10 CFR 52.79(a) for descriptions of operational programs that need to be included in the FSAR in a COL application to support a reasonable assurance finding of acceptability. A COL applicant may rely on information in the applicable Design Certification to help provide a full description of the operational programs for the COL application. At a public meeting on March 26 and 27, 2008, Westinghouse indicated its intent to revise the AP1000 DCD to resolve issues common to COL applicants implementing the AP1000 design. Therefore, the NRC staff reviewed Revision 17 to the AP1000 DCD, Section 3.11, including DCD changes intended to minimize the supplemental information necessary to be provided by COL applicants in fully describing their operational programs in support of the COL applications. As described below for specific review areas, the NRC staff finds the revision to the AP1000 DCD continues to provide an acceptable description of the EQ Program sufficient for the AP1000 Design Certification in accordance with the NRC regulations, pending resolution of the identified confirmatory item in this section.

A COL applicant referencing the AP1000 design will be responsible for fully describing the EQ operational program in support of its COL application. A COL applicant may reference the provisions in the AP1000 DCD as part of its responsibility to fully describe the EQ operational program. The NRC staff will evaluate the full description of the operational EQ Operational Program provided by a COL applicant during review of the COL application consistent with RG 1.206 and NRC SRP Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment."

Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," in AP1000 DCD Tier 2 presents information to demonstrate the mechanical and electrical portions of plant safety systems are capable of performing their designated functions while exposed to applicable normal, abnormal, test, accident, and post-accident environmental conditions. AP1000 DCD Tier 2, Appendix 3D, "Methodology for Qualifying AP1000 Safety-Related Electrical and Mechanical Equipment," describes the methodology to be used to qualify equipment for nuclear power plants with the AP1000 reactor design. During the March 26 and 27, 2008, public meeting, Westinghouse stated procurement specifications were being prepared for safety-related equipment to be used in the AP1000 reactor design. In RAI-SRP3.11-CIB1-01, the NRC staff that requested Westinghouse describe the implementation of the methodology for environmental qualification of safety-related mechanical equipment to be used in the AP1000. In its response to this RAI, in a letter dated May 30, 2008, Westinghouse described its EQ Program for safety-related mechanical equipment. Westinghouse stated safety-related functions of mechanical equipment shall be shown to be acceptable under the required operating conditions and environmental parameters. Further, the AP1000 harsh and mild environmental conditions will be supplied to the vendor in the design and qualification specifications.

On October 14 and 15, 2008, the NRC staff conducted an audit of design and procurement specifications, including environmental qualification, for pumps, valves, and dynamic restraints to be used for the AP1000 reactor at the Westinghouse office in Monroeville, PA. The staff found Westinghouse had included ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," in its design and procurement specifications for AP1000 components, including ASME QME-1-2007, Appendix QR-B, "Guide for Qualification of Nonmetallic Parts." Further, AP1000 DCD Tier 2 (Revision 17), Subsection 5.4.8.3, "Design Evaluation," states the requirements for qualification testing of power-operated active valves are based on ASME Standard QME-1-2007 as listed in AP1000 Tier 2, Subsection 5.4.16, "References." In a memorandum dated November 6, 2008, the NRC staff documented the results of the audit with specific OIs (ADAMS Accession Number ML083110154). In a letter dated January 26, 2010, Westinghouse discussed its plans to address the October 2008 audit findings. Resolution of the audit findings were tracked as Open Item OI-SRP3.11-CIB1-01. In a letter dated February 23, 2010, Westinghouse provided its response to Open Item OI-SRP3.11-CIB1-01. In particular, Westinghouse stated the valve design specifications indicate active valves will be qualified in accordance with ASME Standard QME-1-2007. Westinghouse also provided planned changes to the AP1000 DCD to ensure consistency in the EQ provisions and tables.

As discussed in Section 3.9.6 of this SER, the NRC staff conducted a follow-up audit at the Westinghouse office in Rockville, Maryland, on May 17, 2010, to review the revisions to the design and procurement specifications prepared since the October 2008 audit. Based on the May 2010 audit, the NRC staff found Westinghouse has updated the design and procurement specifications to address NRC comments provided during the October 2008 audit. For example, the staff found the design and procurement specifications require the application of ASME Standard QME-1-2007 for the qualification of mechanical equipment to be used in an AP1000 reactor. Further, the staff found the equipment qualification specification mandates non-metallic components be environmentally qualified using the provisions of Appendix QR-B to ASME Standard QME-1-2007. Based on the follow-up audit, Open Item OI-SRP3.11-CIB1-01 is closed. **Confirmatory Item CI-SRP3.11-CIB1-01** will be used to track the planned AP1000 DCD changes related to the EQ provisions and tables.

Section 3.11.5, "Combined License Information Item for Equipment Qualification File," in Revision 17 to the AP1000 DCD states the COL holder will define the process and procedures for which the equipment qualification files will be accepted from Westinghouse and how the files will be retained and maintained in an auditable format for the period that the equipment is installed and/or stored for future use in the nuclear power plant. In RAI-SRP3.11-CIB1-02, the staff requested Westinghouse to specify the necessary actions for the COL applicant to establish the process and procedures for accepting, maintaining, and storing equipment qualification files. In its response to this RAI in a letter dated May 30, 2008 (ADAMS Accession Number ML081550224), Westinghouse stated it will act as the agent for the COL holder during the equipment design phase, equipment selection and procurement phases, equipment qualification phase, plant construction phase, and ITAAC inspection phases. Westinghouse indicated the COL applicant will provide supplemental information to fully describe the process for retention and maintenance of the EQ documentation for the operational life of the plant. The staff considers the RAI response clarifes the role of Westinghouse in the EQ process; RAI-SRP3.11-CIB1-02, is closed.

The staff reviewed the revisions to the AP1000 DCD with respect to the EQ Program for electrical equipment important to safety and safety-related mechanical equipment. The staff finds the changes are generic and are expected to be applicable to all COL applications referencing the AP1000 certified design.

3.11.2 Conclusion

The NRC staff concludes Revision 17 to the AP1000 DCD continues to satisfy the NRC regulations for electrical and mechanical equipment within the scope of the EQ Program for the AP1000 certified design. The revision to the AP1000 certified design provides sufficient information to satisfy 10 CFR Parts 50 and 52 for the EQ Operational Program for electrical and mechanical equipment to be used at an AP1000 nuclear power plant, pending resolution of the confirmatory item. Further, the staff concludes the AP1000 DCD changes related to the EQ Operational Program are generic and are expected to apply to all COL applications referencing the AP1000 Design Certification.

3.12 Piping Design

The AP1000 DCD, Revision 15 was approved by the staff in the certified design. In the AP1000 DCD Revision 17, the applicant proposed the completion of Design Specification and Reports which is COL Information Item 3.9.2 in Subsections 3.9.3.1.4, 3.9.3.1.5, 3.9.3.4 and 3.9.8.2 and 3.9.8.6 of the DCD. In addition, COL Information Item 3.9-6 would be closed in Section 3.9.8.6. In Appendix 3I, the applicant proposed to address hard rock sites which show higher amplitude at high frequency than the Certified Seismic Design Response Spectra (CSDRS). In Appendix 3C, the applicant proposed to remove the containment interior building structure and the surge line piping from the original reactor coolant loop (RCL) model and provided more accurate description for the RCL model and analysis methods.

In Section 3.9.3.1.2, the applicant revised piping lines connected to the reactor coolant system from not susceptible to thermal stratification, cycling or striping (TASCS) to susceptible to TASCS. The applicant added clarification to Subsections 3.9.3.1.4 and 3.9.3.1.5. The applicant proposed changes to the requirement for the welded connections of ASTM A500 Grade B tube steel members as described in Subsection 3.9.3.4. In Subsection 3.9.3.4, the pipe support deflection limit and pipe support stiffness values used in the piping analysis were clarified. Clarification was added in Subsection 3.9.8.6 to address COL Information Item related to piping benchmark program. Lastly, the applicant proposed changes in Subsection 3.9.8.2 to remove piping DAC from the DCD.

3.12.1 Staff Evaluation

The staff reviewed the proposed changes to the piping design in the AP1000 Revision 17 in accordance with the guidance in the SRP Section 3.12, "ASME Code Class 1, 2, and 3 Piping System, Piping Components and their Associated Supports." The regulatory basis for Section 3.12 of the AP1000 DCD is documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design."

3.12.1.1 Design Specification and Reports

In Subsection 3.9.8.2 the applicant stated "COL holder referencing the AP1000 design will have available for NRC audit the design specifications and as-designed reports prepared for major ASME Section III components and ASME Code, Section III piping." The statement implied the COL applicant may not complete the piping design prior to issuance of a COL.

On February 8, 2008, Westinghouse issued a letter (ADAMS Accession Number ML080440066) related to schedule for piping design document review. In this letter, Westinghouse stated "It is the intention of Westinghouse that design documents related to DAC and COL Information Item will be available for NRC review during the period scheduled for the NRC review of the design certification amendment. It is expected information will be available for NRC review to permit the resolution, closure, or removal of the DAC and COL Information Item."

In RAI-SRP3.12-EMB-4, the staff questioned whether Westinghouse would complete the as-designed piping analyses and design reports by December 2008 as stated in the February 8, 2008, letter. The staff requested Westinghouse revise the DCD to reflect the design completion or propose a method and schedule to resolve the piping DAC issue and update the DCD to reflect the proposed alternative.

By letter dated June 20, 2008 (ADAMS Accession Number ML0817801740), Westinghouse responded it was the intention of Westinghouse to have the design documents for the risk-significant piping packages identified by the NRC available during the review of the design certification amendment. The DCD would be revised to reflect the expected completion of the piping design. It was also expected the NRC's review of these documents would permit the resolution, closure, or removal of the DAC and the COL Information Item. The staff reviewed the applicant's response and its follow-up letter (ADAMS Accession Number ML090220273), which indicated the schedule for the risk-significant piping packages would be completed by June 30, 2009, in order to resolve the piping DAC.

By letter dated April 1, 2010 (ADAMS Accession Number ML100970364), the applicant stated Westinghouse would not be able to complete the piping analysis for the previously specified risk significant piping packages to support the Design Certification Amendment (DCA). In this letter, the applicant revised DCD sections 3.9.8.2 and 3.9.8.7 to address all aspects related to as-designed piping design specification, design reports, and analysis. Additionally, the applicant revised Section 14.3.2.2 of DCD to address DAC/ITAAC closure process for OI-SRP3.12-EMB-04.

Westinghouse proposed the Piping DAC become a COL Information Item (3.9-7) in the DCD for the DCA. COL Information Item 3.9-7 will state the COL applicant needs to complete the as-designed piping analysis for the identified risk significant piping packages to close the piping DAC. The COL Information Item may be addressed by the COL applicant in a manner that complies with NRC guidance provided in RG 1.215, and outlined in Appendix 14.3A of the DCD. The applicant's use of piping DAC was previously approved and certified in Revision 15 of the DCD and documented in NUREG-1793. The staff finds this acceptable.

By letter dated August 23, 2010 (ADAMS Accession Number ML102380040), the applicant provided a Revision 2 response to OI-SRP3.12-EMB-04 to address RAI-SRP3.12-EMB1-09 that is documented in Section 3.12.1.3 of this SER. In this letter, the applicant revised Sections 3.9.8.2 and 3.9.8.7 to further clarify as-designed design specifications, design reports, and

analysis will be the responsibility of the COL applicants. The applicant also clarified the availability of the piping design information and design reports will be identified to the NRC. The staff has determined these clarifications for the COL applicants' activities are acceptable. In that letter, the applicant added a Table 3.9-20, which describes piping packages chosen to demonstrate piping design for piping DAC closure (in addition to Class 1 lines larger than one inch in diameter). The staff reviewed this Table and concluded these piping packages and Class 1 lines do represent the Class 1, 2, and 3 piping packages and would be able to demonstrate piping design for piping DAC closure. Therefore, the staff finds this acceptable.

In this letter, the applicant also provided a markup of Section 14.3A of the DCD for DAC/ITAAC closure process as follows:

> 14.3A Design Acceptance Criteria / ITAAC Closure Process
>
> DAC (Design Acceptance Criteria) are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification. DAC are to be objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as a part of the ITAAC performed to demonstrate the as-built facility conforms to the certified design. (SECY 92-053)
>
> There are three process options for DAC / ITAAC resolution:
>
> - Resolve through amendment to design certification
> - Resolve as part of COL review
> - Resolve after COL is issued
>
> In the first two options, the applicant will submit the design information and the NRC will document its review in a safety evaluation. In the third option, the COL holder notifies the NRC of availability of design information and the staff will document its review in an inspection report.
>
> Should the third option be implemented for the first standard AP1000 plant, subsequent COL applicants may reference the first standard plant closure documentation and close the DAC / ITAAC under the concept of "one issue, one review, one position," identified in NRC guidance.
>
> Additionally, Westinghouse Electric Company may submit licensing topical reports for NRC review of the material supporting the DAC / ITAAC closure and request the NRC to issue a safety evaluation in conjunction with a closure letter or inspection report concluding the acceptance criteria of the DAC / ITAAC have been met. Subsequent COL applicants may reference these reports and NRC closure documents in effort to close the DAC / ITAAC.
>
> For technical areas where DAC / ITAAC apply in the design certification rule, COL applicants will provide an ITAAC and associated closure schedule indicating the approach to be applied.

For subsequent COL applicants following the first standard AP1000 plant, the application could reference the existing DAC / ITAAC closure documentation for the first standard plant.

NRC guidance for DAC/ITAAC is provided in RG 1.206, Section C.III.5. Further information on the staff's position of DAC/ITAAC being used as part of the 10 CFR Part 52 review process is provided in SECY-92-053.

The staff reviewed and evaluated the applicant's letter for addressing OI-SRP3.12-EMB-4. On the basis that the applicant-proposed position provides no change from the approved design (the use of Piping DAC was approved) and better defines a plan for closure later in the construction period, the staff finds this acceptable and OI-SRP3.12-EMB-4 is resolved pending Westinghouse revision of the DCD. This item is being tracked as **Confirmatory Item CI-SRP3.12-EMB-4.**

In RAI-SRP3.12-EMB-5, the staff questioned how the COL holder would complete verification of the thermal cycling and stratification loading considered in the stress analysis as discussed in Subsection 3.9.3.1.2 prior to fuel load. The staff was not aware of a specific monitoring program for verification of thermal cycling and stratification loading condition of the automatic depressurization Stage 4 lines and the passive residual heat removal line. These two lines are susceptible to thermal stratification as described in Subsection 3.9.3.1.2 of DCD. If verification could not be completed prior to fuel load, the applicant was requested to provide alternatives.

By letter dated June 20, 2008 (ADAMS Accession Number ML0817801740), the applicant responded Subsection 3.9.8.2 deals with design specifications and design reports and the requirement to perform a reconciliation/analysis for the as-built piping. The intent of the phrase in parenthesis (verification of the thermal cycling and stratification loading considered in the stress analysis discussed in Section 3.9.3.1.2) was to verify "dimensional/layout/support differences" identified in an as-built walk-down were considered in thermal cycling/stratification as well as the standard portion of the piping analysis. The monitoring program identified in Subsection 3.9.8.5 was a one-time requirement for the surge line and was not related or applicable to Subsection 3.9.8.2. Thermal cycling and stratification loading were to be evaluated by analysis and if the as-built dimensions, layout, or supports on the piping lines changed as the result of construction, a reconciliation of the stratification analysis was to be performed. The staff reviewed the clarification provided in the response and determined it was acceptable.

3.12.1.2 Closure of COL Information Item 3.9-6 (Piping Benchmark Program)

The original COL Information Item commitment stated the COL applicant will implement a benchmark program as described in Subsection 3.9.1.2 if a piping computer program other than the one used for design certification is used. The piping benchmark problems identified in Reference 20 for the Westinghouse AP600 are also representative of AP1000 and can be used for the AP1000 piping benchmark program if required.

In Subsection 3.9.8.6, the applicant proposed to close out the COL Information Item 3.9-6. The applicant stated the combined license information requested in this subsection had been completely addressed in TR-15, "Benchmark Program for Piping Analysis Computer Programs,"

APP-GW-GLR-006, March 2006, and no additional work was required by the combined license applicant to address the combined license information requested in this subsection.

The staff reviewed TR-006, "AP1000 As-Built COL Information Items," APP-GW-GLR-021, which stated all piping analysis performed for AP1000 was being completed using only programs that had already been benchmarked to NRC's satisfaction. PIPESTRESS, GAPPIPE, WECAN and ANSYS require no additional benchmarking by the COL Applicant. On the basis that the above mentioned computer codes have been accepted by the staff and other analysis codes are not being used for piping analysis, the staff finds this change acceptable and COL Information Item 3.9.6 is closed.

3.12.1.3 Evaluation for High Frequency Seismic Input

The staff reviewed the applicant's Appendix 3I of DCD Revision 17, and its supporting document, TR-115, "Effects of High Frequency Seismic Content on SSCs," APP-GW-GLR-115, October 2007. However, the seismic input has been identified in Section 3.7.3 as inadequate due to mathematical model error. The applicant revised TR-115 with adequate seismic input. The staff reviewed the revised document, TR-115, Revision 2.

TR-115, Revision 2 states the HRHF exceeds the CSDRS for frequencies above about 15 HZ and the representative piping considered is based on high frequency participation as indicated in Section 6.3.1 of TR-115.

The staff noted the floor response spectra exceedances were in the region for frequencies around 6-12 Hz as shown in Figure 6.3.2.2-1 through 6.3.2.2-3 of TR-115. The staff identified the selected representative piping systems only addressed high frequency piping systems. TR-115 failed to address all other piping packages related to the floor response spectra exceedances for frequencies around 6-12 HZ. The staff reviewed sample piping system selection criteria and determined the selection criteria was not adequate. The staff also noted TR-115, Revision 2 did not address support load increase due to the response spectra exceedance. The staff issued OI-SRP3.12-EMB1-09 to ask applicant to address all piping packages for the floor response spectra exceedances around 6-12 HZ and piping support load increases.

By letter dated August 17, 2010 (ADAMS Accession Number ML102350447), the applicant provided the following response:

> "HRHF GMRS effects on ASME Class 1, 2 and 3 piping systems will be evaluated as part of the Piping DAC, captured as COL Information Item 3.9-7 in the DCD (See response to Open Item OI-SRP3.12-EMB-4 R1 in Letter No. DCP_NRC_002845).
>
> Areas of exceedance of the CSDRS will be addressed for the entire frequency range.
>
> The following will be considered in the evaluation:
>
> - Piping Qualification
> - Support Loads
> - Valve Accelerations
> - Valve End Stresses

- Equipment Nozzle Loads

Impacts on the following evaluations will be considered:

- Pipe Rupture Hazards Analysis
- LBB
- Piping and Component Fatigue Analysis"

The applicant also provided a markup of revised Table 3.9-19 and revised Appendix 3I to state ASME Class 1, 2 and 3 piping systems will be evaluated for the HRHF GMRS and this evaluation is within the scope of the piping DAC.

The staff reviewed the applicant's response and identified COL Information Item 3.9-7 is not listed in the Table 1.8-2 of DCD. The staff requested the applicant to address COL Information Item 3.9-7(as-designed piping analysis). By letter dated August 23, 2010 (ADAMS Accession Number ML102380040), the applicant provided a markup of revised Table 1.8-2 by adding this COL Information Item to address staff's concern. In this letter, the applicant also revised its DAC/ITAAC closure process. The evaluation of DAC/ITAAC closure process is documented in Section 3.12.1.1.

In the markup of revised Appendix 3I of DCD, the applicant stated the COL will evaluate the HRHF GMRS effects of the piping systems by changing the piping system screening criteria from at least 2 piping analysis package to all ASME Class 1, 2, and 3 piping systems. The staff determined the revised position meets design requirements of GDC 2. Therefore, the staff finds this acceptable. This is being tracked as **Confirmatory Item CI-SRP3.12-EMB1-09**, pending Westinghouse revision of the DCD.

3.12.1.4 Reactor Coolant Loop Analysis Methods

The staff reviewed the proposed change in Appendix 3C of DCD Revision 17, and its supporting document TR-13 Revision 1, "Safety Class Piping Design Specifications and Design Reports Summary," APP-GW-GLR-013, May 2007. The proposed change would remove the containment interior building structure and the surge line piping from the reactor coolant loop model description. The staff also reviewed the applicant's proposed change related to time history analysis in Subsection 3.9.3.1.4 of DCD Revision 17. The applicant clarified that unless appropriate time-history seismic input from the building is provided at multiple supported locations, the containment internals structure is included in the system-coupled model in the time-history analysis. The staff agreed that a containment interior building structure is not required because the seismic inputs to the RCL model are provided at all of the building attachments to the Reactor Coolant Loop (RCL).

TR-13 identifies that pressurizer surge-line piping is to be analyzed in APP-RCS-PLR-040 as listed in Table 2. In RAI-SRP3.12-EMB-6, the staff noted that the reactor coolant loop analysis did not couple the branch lines such as the pressurizer surge line. Subsection 3.7.3.8.1 of the DCD states that "if the ratio of the run piping outside diameter to the branch piping outside diameter (nominal pipe side) exceeds or equals 3.0, the branch piping can be excluded from the analysis of the run piping." Several branch lines do not meet this ratio, and therefore, should be included in the RCL piping analysis. The staff requested that the applicant explain this discrepancy and take action to address this DCD conformance issue.

By letter dated December 23, 2008 (ADAMS Accession Number ML083650184), the applicant responded as follow:

> The branch piping has been excluded from the reactor coolant loop analysis because the criteria in Subsection 3.7.3.8.1 do not apply to the hot and cold leg piping. Just as attached piping is excluded from primary equipment models, the branch piping of the surge line, automatic depressurization system Stage 4 (ADS4), RNS suction line, and several smaller lines are excluded from the analysis of the hot and cold leg piping.
>
> The reactor coolant loop (APP-RCS-PLA-050) is unique in that the stiffness and mass characteristics are closer to that of equipment than a typical piping analysis package. With the relatively short run length, comparatively large pipe diameter and pipe thickness, both the cold and hot legs have much less flexibility than a typical run length of pipe. The large interplay of the hot and cold leg piping with the reactor pressure vessel and steam generator extends the boundary of the piping analysis package to include primary equipment as well as primary loop piping.
>
> "No non-conformance exists because the reactor coolant loop piping is treated as a rigid piece of equipment (fundamental frequency greater than 33 Hz) and not a flexible pipe.

In this letter, the applicant also submitted the DCD revision for Subsection 3.7.3.8.1 and Appendix 3C to reflect that the branch piping is excluded from the reactor coolant loop analysis.

The staff reviewed the RCL layout configuration, which showed a total length of the 37.5" outside diameter hot leg pipe to be approximately 20 ft, which should conform to a rigid body motion. On the basis that the applicant has performed a calculation to demonstrate that AP1000 RCL piping is rigid and has a fundamental frequency much higher than 33 Hz, the staff finds this acceptable.

On the basis of above discussion, the staff found that the proposed change is acceptable to reflect the RCL model used in the loop analysis.

3.12.1.5 Remove Piping Design Acceptance Criteria (DAC)

The staff reviewed the proposed changes in the introduction of DCD Revision 17 and related Tier 2 Subsection 3.9.8.2 and Table 3.9-19. The staff determined that risk-significant piping design packages would have to be completed in order to resolve or remove reference to piping DAC.

DCD Subsection 3.9.8.2 was revised to reflect the design completion by indicating that as-designed design specifications and design report for the major ASME Code, Section III components and piping are available for NRC review.

During the period October 20 - 24, 2008, the staff performed an on-site review, at the Westinghouse headquarters, of AP1000 ASME Class 1 piping and support design with the intent to resolve piping DAC. During this review, the staff found that the applicant had not completed risk-significant ASME Class 1 piping analysis packages. On the basis that the risk-significant piping analyses had not been completed, the staff cannot remove piping DAC at this

time. The on-site review summary is documented in a letter dated December 30, 2008 (ADAMS Accession Number ML083500308).

By letter dated January 19, 2009 (ADAMS Accession Number ML090220273), the applicant submitted AP1000 piping DAC Analysis Schedule. In this letter, the applicant stated that the AP1000 ASME Code, Section III, Class 1, 2, and 3 piping analysis packages are rescheduled to be completed by June 30, 2009. The applicant will inform the staff when it is ready for another on-site review for Class 1, 2, and 3 risk-significant piping analysis to complete resolution of the piping DAC. This is Open Item OI-SRP3.12-EMB-4.

By letter dated April 1, 2010 (ADAMS Accession Number ML100970364), the applicant stated that Westinghouse would not be able to complete the piping analysis for the previously specified risk significant piping packages to support the Design Certification Amendment (DCA). The resolution and evaluation are discussed and evaluated in Section 3.12.1.1 of this report.

3.12.1.6 Change Component and Piping Support Weld Connections Requirement

The staff reviewed the proposed changes in Tier 2 Subsection 3.9.3.4 of DCD Revision 17. Section 3.9.3.4 stated that the welded connections of ASTM 500 Grade B tube steel members satisfy the requirements of the AISC "Load and Resistance Factor Design (LRFD) Specification for Steel Hollow Structure Sections," dated November 10, 2000. SRP 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," SRP Acceptance Criteria 2, identified applicable steel structure codes, standard, and specifications. The applicant proposed LRFD Specification is not listed as acceptable. SRP proposed "ANSI/AISC N690-1994 including Supplement 2 (2004)" as an acceptable specification. ANSI/AISC N690-1994 including Supplement 2 (2004) has been accepted by the NRC as ASME Code Case N-570-2. The later LRFD version of AISC N690, ASME Code Case N-721, has not been accepted by the NRC. The staff noted that the NRC's current acceptable specification is based on Allowable Stress Design (ASD) specification. Further, the LRFD method has not been approved for use in the design of new reactor nuclear facilities. In RAI-SRP3.12-EMB-8, the staff requested the applicant to identify differences between the two methods and show equivalency with respect to SRP acceptable specification or provide alternatives to satisfy the acceptance criteria.

In a letter dated July 30, 2009 (ADAMS Accession Number ML092150355), the applicant stated that the AP1000 component and piping support designs satisfy the requirements of the ASME Code Section III, Subsection NF and the requirements in the DCD on the welding of members fabricated on tube steel are in addition to the requirements in Subsection NF. These requirements are not considered to be an alternative to the Subsection NF requirements. On the basis that the applicant meets the requirements of ASME Section III, Subsection NF, any additional requirements imposed by the applicant shall provide additional level of quality and safety. Therefore, the staff finds this acceptable. The applicant stated that it would revise the DCD to clarify this issue. The staff will review the next revision of the DCD to confirm this clarification. This is **Confirmatory Item CI-SRP3.12-EMB-8**.

3.12.1.7 Revision of RCS Lines from Not Susceptible to TASCS to Susceptible to TASCS

The staff reviewed the applicant's proposed change related to piping lines susceptible to TASCS in Section 3.9.3.1.2 of DCD Revision 17. The staff reviewed piping and instrument

drawings for these lines identified by the applicant and determined these lines are susceptible to TASCS. Therefore, the staff finds this acceptable.

3.12.1.8 Piping Design Methods

The staff reviewed the applicant's proposed change related to piping design methods and criteria in Section 3.9.3.1.5 of DCD Revision 17. The applicant summarized the methods and criteria used in design and analysis of the ASME Code Classes 1, 2, and 3 in Table 3.9-19. The staff reviewed Table 3.9-19 and determined that the applicant's summarization is acceptable.

3.12.1.9 Pipe support deflection limit and pipe support stiffness

The staff reviewed the applicant's proposed change related to pipe support design in Section 3.9.3.4 of DCD Revision 17. The applicant's change from dynamic loading to dynamic combined faulted loading is for clarification and considered an editorial change. The editorial change for support stiffness also provides clarification. The staff finds these editorial changes for clarification acceptable.

3.12.2 Conclusion

Based on its review of the information provided in the AP1000 Amendment up to Revision 17, the staff concludes that supports of piping systems important to safety are designed to quality standards commensurate with their importance to safety. Section 3.12.1.1 of this SE discusses the path to completion of piping analysis, including piping supports. The staff also concludes that the applicant satisfies the following:

- The requirements of GDC 1 and 10 CFR 50.55a by specifying methods and procedures for the design and construction of safety-related pipe supports in conformance with general engineering practice.

- The requirements of GDC 2 and 4 by designing and constructing the safety-related pipe supports to withstand the effects of normal operation, as well as postulated events such as LOCAs and dynamic effects resulting from the SSE.

- 10 CFR Part 50 requirements by identifying applicable codes and standards, design and analysis methods, design transients and load combinations, and design limits and service conditions to assure adequate design of all safety-related piping and pipe supports in the AP600 for their safety functions.

- 10 CFR Part 52 requirements by providing reasonable assurance that the piping systems will be designed and built in accordance with the certified design. Through the performance of the ITAAC, the COL holder will verify the implementation of these preapproved methods and satisfaction of the acceptance criteria. This will assure that the as-constructed piping systems conform to the certified design for their safety functions.

- 10 CFR Part 50, Appendix S, requirements by designing the safety-related piping systems with a reasonable assurance that they will withstand the dynamic effects of

earthquakes with an appropriate combination of other loads of normal operation and postulated events with an adequate margin for ensuring their safety functions.

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