AP1000DCDFileNPEm Resource

From: Butler, Rhonda
Sent: Wednesday, July 28, 2010 8:08 AM
To: AP1000DCDFileNPEm Resource
Subject: FW: AP1000 Response to RAI on SRP3.9.4. 07/15/2010
Attachments: Chapter 3 P4 SER Huang 2010July.docx; image001.jpg

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From: Huang, Jason
Sent: Tuesday, July 27, 2010 2:35 PM
To: Clark, Phyllis; Dixon-Herrity, Jennifer; Gleaves, Bill
Cc: Spicher, Terri; McKenna, Eileen; Hsia, Anthony; Chen, Pei-Ying
Subject: RE: AP1000 Response to RAI on SRP3.9.4. 07/15/2010

Phyllis,

Attached is the revised Section 3.9.4 SER writeup, with new section 3.9.4.1.3 Seismic Qualification of CRDM.

Jason Huang
General Engineer
NRO/DE/EMB1
301-415-2974

From: Clark, Phyllis
Sent: Monday, July 26, 2010 10:40 AM
To: Dixon-Herrity, Jennifer; Gleaves, Bill
Cc: Spicher, Terri; McKenna, Eileen; Hsia, Anthony; Huang, Jason; Chen, Pei-Ying
Subject: RE: AP1000 Response to RAI on SRP3.9.4. 07/15/2010

Hi Jen,

Please let me know when we can expect the Section 3.9.4 revised SER write-up.

Thanks,

P. Clark
Project Manager
U.S. Nuclear Regulatory Commission
Office of New Reactors, DNRL/NWE2
Room T-6C10
Washington, DC 20555
301-415-6447
Phyllis.Clark@nrc.gov

From: Dixon-Herrity, Jennifer
Sent: Wednesday, July 21, 2010 9:51 AM
To: Gleaves, Bill
Cc: Clark, Phyllis; Spicher, Terri; McKenna, Eileen; Hsia, Anthony; Huang, Jason; Chen, Pei-Ying
Subject: RE: AP1000 Response to RAI on SRP3.9.4. 07/15/2010

Actually, we are in the process of modifying the SE writeup. This issue will be documented to ensure that the outcome is recorded for the future.

Jen

From: Gleaves, Bill
Sent: Wednesday, July 21, 2010 9:35 AM
To: Dixon-Herrity, Jennifer
Cc: Clark, Phyllis; Spicher, Terri
Subject: FW: AP1000 Response to RAI on SRP3.9.4. 07/15/2010

Jennifer,

Please let us know if the concern is resolved. We don't expect any updated SER input. Let us know if you have SER changes.

Billy

William (Billy) Gleaves
US NRC/NRO/DNRL/NWE2
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From: McKenna, Eileen
Sent: Monday, July 19, 2010 12:17 PM
To: Gleaves, Bill
Cc: Clark, Phyllis; Spicher, Terri; Butler, Rhonda
Subject: RE: AP1000 Response to Request for Additional Information (SRP 3). 07/15/2010

The RAI response goes to Jennifer's branch. Note that the questions for 3.9.4 were on Rev. 15 information. There are no DCD changes, there was no OI in the phase 2 SER; thus, we don't need any SE input, only agreement from the tech branch that the concerns are resolved.

From: Gleaves, Bill
Sent: Monday, July 19, 2010 12:11 PM
To: McKenna, Eileen
Cc: Clark, Phyllis; Spicher, Terri; Butler, Rhonda
Subject: FW: AP1000 Response to Request for Additional Information (SRP 3). 07/15/2010

Response to RAI-SRP3.9.4-EMB 1-02. Eileen, what should we do with this response? Is this a confirmatory item for closure?

Billy

William (Billy) Gleaves
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From: E-RIDS2 Resource
Sent: Monday, July 19, 2010 11:30 AM
To: Bajorek, Stephen; Buckberg, Perry; Gleaves, Bill; Lui, Christiana; Proctor, Christopher; RidsAcrsAcnw_MailCTR Resource; RidsManager Resource; RidsNroDnrlNwe2 Resource; RidsNroPMDJaffe Resource; RidsNroPMMMiernicki Resource; RidsNrrDe Resource; RidsNrrDirsIolb Resource; RidsNrrDlrRer1 Resource; RidsNrrDorl Resource; RidsOcfoMailCenter Resource
Subject: AP1000 Response to Request for Additional Information (SRP 3). 07/15/2010

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Title	AP1000 Response to Request for Additional Information (SRP 3).
Docket Number	05200006
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Author Name	Sisk R
Author Affiliation	Westinghouse Electric Co
Addressee Name	
Addressee Affiliation	NRC/Document Control Desk NRC/NRO
Document Type	Letter
Availability	Publicly Available
Date to be Released	07/27/2010
Document Sensitivity	Non-Sensitive
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Subject: FW: AP1000 Response to RAI on SRP3.9.4. 07/15/2010
Sent Date: 7/28/2010 8:07:49 AM
Received Date: 7/28/2010 8:07:52 AM
From: Butler, Rhonda

Created By: Rhonda.Butler@nrc.gov

Recipients:
"AP1000DCDFileNPEm Resource" <AP1000DCDFileNPEm.Resource@nrc.gov>
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3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

Revision 17 of the DCD includes a number of changes to Subsection 3.2.1, "Seismic Classification," Tables 3.2-2, "Seismic Classification of Building Structures" and 3.2-3, "Seismic Classification of Mechanical and Fluid Systems, Components, and Equipment." The change to Subsection 3.2.1.1.1 is limited to a clarification regarding reference to 10 CFR 50.34 rather than 10 CFR 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

The staff reviewed the proposed changes to the DCD according to the guidance in the AP1000 NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," SRP 3.2.1, "Seismic Classification," which references Regulatory Guides (RGs) 1.29, Revision 3, "Seismic Design Classification"; RG 1.143, Revision 2, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants"; RG 1.151, Revision 0, "Instrument Sensing Lines"; and RG 1.189, Revision 0, "Fire Protection for Operating Nuclear Power Plants for Seismic Classification of Various Structure, 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. The staff review considered additional detailed design information needed to be verified under the Design Acceptance Criteria (DAC). A DCD Section 3.9.3, "ASME Code Classes 1, 2, and 3 Components, Component Supports, and Core Support Structures" NRC audit of design specifications performed October 13, 2008, through October 17, 2008[1], for risk-significant components was also considered relative to seismic classification. The staff reviewed related Technical Reports (TRs) and also reviewed the non-site-specific SSCs included in DCD Subsection 3.2.1 to determine if the scope was essentially complete.

The staff determined that the proposed DCD Subsection 3.2.1 change referencing 10 CFR 50.34 rather than 10 CFR 100 was acceptable, since 10 CFR 50.34 is referenced in the definition of the term safety-related in addition to 10 CFR 100. Both regulations provide similar acceptance criteria for off-site doses. The other DCD changes were primarily intended to resolve the staff's questions on the regulatory treatment of non-safety systems (RTNSS). The staff also 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

[1] The results of the October 17, 2008 Audit are documented in an August 3, 2009 Audit Report "Regulatory Audit Report AP1000 Design Certification Amendment Section 3.9.3 on October 13 – 17, 2008" (Agency wide Document Access and Management System Accession Number ML092150664).

16 the staff identified several potential errors and omissions in a number of technical areas that need clarification in the DCD. The staff reviewed Revision 17 to determine if the open items identified during the Revision 16 review could be closed. The staff's review evaluated the DCD changes to determine if they were appropriate to resolve these errors and omissions and these are discussed below under each topic. The technical review and resulting Requests 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.

3.2.1.1.1 Augmented Seismic Requirements for RTNSS SSCs

To comply with 10 CFR 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 consideration under the RTNSS process to enable them to withstand earthquakes and meet GDC 1. The extent that nonsafety-related SSCs are seismically qualified is defined by the RTNSS process.

In Revision 17 to the DCD, 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 an RAI response defining additional seismic requirements for this RTNSS equipment to be located within buildings designed to Uniform Building Code 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 Commission's memorandum dated June 23, 1997, concerning implementation of the staff position in SECY-96-128: RTNSS equipment required post-72 hours needs to be located such that there are no special interactions with any other non-seismic SSCs. On the basis of the Commission's memorandum, no dynamic qualification of active equipment is necessary for RTNSS SSCs and the 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 classifications of SSCs are 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 non-seismic, comments in the table stipulate a seismic analysis consistent with the American Society of Mechanical Engineers (ASME) Section III Class 3 systems. The staff finds this to be acceptable, since this meets the criteria for seismic analysis identified in SRP 9.5.1 and RG 1.189 for portions of fire protection systems.

With regard to additional seismic requirements that may apply to certain Class D systems and components, it is still not clear what those additional requirements are. DCD Subsection 3.2.2.6 states that, with 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. The 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 staff memorandum 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. The staff is concerned that seismic anchorage alone does not ensure functionality of electrical and mechanical equipment following an SSE, unless it is supported by an analysis or experience. This concern is identified as **Open Item OI-SRP3.2.1-EMB2-01**.

In an attempt to resolve this Revision 16 Open Item, the staff performed an on-site review to examine detailed design documents that could define the additional information for staff to reach a safety conclusion. The results of the on-site review are documented in the NRC report dated March 17, 2009, (ADAMS Accession Number ML090640247) "NRC On-Site Review of the Integration of RTNSS with Classification Process and Chapter 19 FSER Open Items in AP1000," and a future audit opportunity may be used to evaluate the additional information requested.

3.2.1.1.2 Scope of SSCs Identified in DCD Subsection 3.2.1

During the review of Revision 16, the 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 Open Item. 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 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 Circulation Water System (CWS) and Raw Water System (RWS) as non-seismic.

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 referenced 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 RPV insulation. This concern was identified during the review of Revision 16 and continues to be identified as **Open Item OI-SRP3.2.1-EMB2-02**.

In an attempt to resolve this Open Item, the staff performed an on-site review to examine detailed design documents that could define the additional information for the staff to reach a reasonable safety conclusion. The results of the on-site review are documented in the NRC report dated March 17, 2009, "NRC On-Site Review of the Integration of RTNSS with Classification Process and Chapter 19 FSER Open Items in AP1000," and a future audit opportunity may be used to evaluate the additional information requested.

3.2.1.1.3 Augmented Quality Assurance Requirements for Seismic Category II SSCs

In Revision 16 of the DCD Subsection 3.2.1.1.2 was revised to reference DCD Section 17.5, "Combined License Information Items," rather than 17.4, "Design Reliability Assurance Program," for the combined license Quality Assurance (QA) requirements for seismic Category II SSCs. During the review of Revision 16, the 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 DCD Section 3.2, DCD Table 3.2-3 or DCD Chapter 17 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, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," 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 to the DCD 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 is identified as **Open Item OI-SRP3.2.1-EMB2-03**.

In an attempt to resolve OI-SRP3.2.1-EMB2-03, the staff performed an on-site review to examine detailed design documents that could define the additional information for the staff to reach a reasonable safety conclusion. The results of the on-site review are documented in the NRC report dated March 17, 2009, "NRC On-Site Review of the Integration of RTNSS with Classification Process and Chapter 19 FSER Open Items in AP1000" and a future audit opportunity may be used to evaluate the additional information requested.

3.2.1.1.4 List of SSCs Needed for Continued Plant Operation

10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," Section 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 3.2.1 states that, if the applicant has set the OBE Ground Motion to the value one-third of the Safe Shutdown Earthquake (SSE) Ground Motion, then the applicant should also provide a list of SSCs necessary for continued operation that must remain functional without undue risk of the health and safety of the public and within applicable stress, strain and deformation, 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

United States Nuclear Regulatory 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 that the licensing basis is maintained. Listing the SSCs in the DCD will allow the plant to address the requirements when the need exists.

In an attempt to obtain this information, the staff performed an on-site review to examine detailed design documents that could define the additional information for the staff to reach a reasonable safety conclusion. The results of the on-site review are documented in the NRC report dated March 17, 2009, “NRC On-Site Review of the Integration of RTNSS with Classification Process and Chapter 19 FSER Open Items in AP1000,” and additional information is needed for the staff to make a reasonable safety conclusion.

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. If DCD Table 3.2-3 serves this purpose, the applicant was requested to clearly state in the DCD that the table contains the list of SSCs necessary for continued operation. This concern is identified as **Open Item OI-SRP3.2.1-EMB2-06.**

3.2.1.2 Conclusion

Except for the identified technical issues identified above that are not adequately addressed in the DCD Revision 17 Amendment and require additional information from the applicant, the seismic classification of SSCs is consistent with RG 1.29, with the exceptions identified in DCD Appendix 1A.

Therefore, on the basis of its review of DCD Revision 17 revisions 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

Revision 17 of the DCD includes a number of changes to Subsection 3.2.2, “AP1000 Classification System” and Table 3.2-3, “AP1000 Classification of Mechanical and Fluid Systems, Components and Equipment,” with regard to the AP1000 classification system. The changes to DCD Subsection 3.2.2 include a clarification regarding reference to 10 CFR 50.34 rather than 10 CFR 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 DCD Revisions 17 according to the guidance in the NUREG-0800 SRP 3.2.2, Quality Group Classification that references RG 1.26 for quality group classification of various SSCs. The staff review considered additional detailed design information needed to be verified under DAC. A Section 3.9.3 NRC audit of design specifications performed October 13, 2008 thru October 17, 2008 for risk-significant components was also considered relative to Quality Group classification. The staff also reviewed Technical Report (TR)103, “Fluid System

Changes,” Revision 2, September 2007, and TR-106, “AP1000 Licensing Basis for Mechanical System and Component Design Updates,” June 2007, which address various system changes that could have an impact on quality group classifications:

The staff determined that the DCD Subsection 3.2.2.1 change referencing 10 CFR 50.34 rather than 10 CFR 100 was acceptable since 10 CFR 50.34 is referenced in the definition of the term safety-related in addition to 10 CFR 100. Both regulations provide similar acceptance criteria for off-site doses. The other DCD changes were primarily intended to resolve the staff’s questions on the RTNSS. The staff also determined that the clarifying notes concerning applicability of ASME Section III to pressure boundary components 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 are acceptable, but the staff identified several potential errors and omissions in a number of technical areas that 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.

3.2.2.1.1 Supplemental Requirements for Nonsafety-Related Passive SSCs Important to Safety

During the review of Revision 16 of the DCD, the staff was concerned that DCD Section 3.2 (or Table 3.2-3) does not adequately define specific supplemental quality standards and aspects of the QA program 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 what portions of the 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 Probabilistic Risk Assessment (PRA) did not identify SSCs that need a more rigorous code or Standard than 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, but not entirely, resolves the staff’s concerns. Although the PRA and RTNSS process did not apparently identify any supplemental requirements for passive components, the staff is concerned that supplemental requirements may be appropriate, especially where there is an insufficient operating history. For example, where high density polyethylene (HDPE) piping is to be used for underground plant service water 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 supplemental requirements 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 does have an importance 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, the staff believes that passive SSCs used in risk-significant RTNSS systems such as the PSWS piping should have supplemental requirements applied. This concern is identified as **Open Item OI-SRP3.2.2-EMB2-01**.

In an attempt to resolve the Revision 16 open item, the staff performed an on-site review to examine detailed design documents that could define the additional information for the staff to reach a reasonable safety conclusion. The results of the on-site review are documented in the NRC report dated March 17, 2009, and a future audit opportunity may be used to evaluate the additional information requested.

3.2.2.1.2 Application of Unendorsed ANS Standard

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 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 or Topical Report.

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 until the standard is updated to reflect new passive reactor designs and is NRC endorsed. This staff concern is considered closed.

3.2.2.1.3 Codes and Standards

A Staff Requirements Memorandum 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 DCD 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 standards may be reviewed on a case-by-case basis.

In the September 5, 2008, response to the RAI, 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 agrees that editions of codes and standards in effect six-months prior to the application and not otherwise NRC endorsed, are in general acceptable. The staff will have the opportunity to review specific editions of codes and standards on a case-by-case basis as they are included in the DCD revisions or during audits. DCD Section 3.2.6 Revision 17 made no changes to the referenced codes and standards editions and this concern is considered closed.

3.2.2.1.4 Classification of Fire Protection System (FPS)

During the DCD Revision 16 review the staff was concerned that Subsection 3.2.2.7 had been revised to identify that both Class F and G are used for 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 dated September 5, 2008, the applicant submitted a revised Table 3.2-3 for additional FPS piping and components. DCD Revision 17 includes these FPS classes in Table 3.2-3. The staff concludes 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 FPS is basically consistent with RG 1.29 and the SRP 9.5.1 criteria and is therefore acceptable. The classification of the Standpipe system as AP1000 Class F constructed to ANSI B31.1 and NS with a seismic analysis consistent with ASME Section III Class 3 is basically consistent with the guidance in “Fire Protection System” SRP 9.5.1 and RG 1.189 (considered not applicable to AP1000 in the DCD) and is, therefore, an acceptable regulatory basis. The staff concern with completeness of the FPS classification is considered closed.

3.2.2.2 Conclusion

Except for the technical issue identified above that is not adequately addressed in the DCD Revision 17 and requires additional information from the applicant, the equipment class/quality group classification of SSCs is, in general, consistent with RG 1.26, with the exceptions identified in DCD Appendix 1A.

On the basis of its review of the applicable information in the AP1000 DCD, and the above discussion, the staff concludes that the Quality Group classifications of the pressure-retaining and non-pressure retaining SSCs important to safety, as identified in DCD Tier 2, Tables 3.2-1 and 3.2-3, and related diagrams in the DCD, are in general consistent with RG 1.26 (with acceptable exceptions) and, therefore, are acceptable. These tables and diagrams identify major components in fluid systems (i.e., pressure vessels, heat exchangers, storage tanks, piping, pumps, valves, and applicable supports) and in mechanical systems (i.e., cranes, fuel handling machines, and other miscellaneous handling equipment). In addition, diagrams 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 provides assurance that component quality will be commensurate with the importance of the safety functions of these systems. This constitutes the basis for satisfying 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 1 and is, therefore, acceptable.

3.3 Wind and Tornado Loadings

3.3.1 Summary of Technical Information

With regard to tornado loads on the Passive Containment Cooling System air baffle, the AP1000 DCD Revision 17 changes the geometry of the shield building roof by reducing the roof rise from 25 ft-6 in. (7 m-77 cm) down to 20 ft-6 in. (6 m-25 cm). As a result, the tornado loads carried by the passive containment system air baffle 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. The development of loads on the air baffle in the top portion of the shield building due to the design wind and tornado is a safety concern. The methodology for load evaluation follows the AP600 approach combined with wind tunnel testing, which give 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 changes the roof geometry by reducing the roof rise from 25’ 6” (7 m-77 cm) to 20’ 6”(6 m-25 cm), thereby reducing the total height of the air baffle by 5 feet (1 m-52 cm). 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 (1-1/2 meters) in height from 25½ ft (7-3/4 meters) to 20½ ft (6-1/4 meters) will result in a 20 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 proposed change is favorable in terms of wind loading only and in no way implies that the change is also favorable to other design parameters. There appears to be one more extra benefit derived from this design change, namely the risk of the building’s being hit by foreign objects, either man-made or natural, is reduced because the area exposed to the hazards is less than before the change. The result is increased safety and security.

3.3.4 Development of COL Information Items

The AP1000 DCD Revision 15 (i.e., NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design”) requires closure of COL Information Items by the COL applicant in Subsection 3.3.3, “Combined License Information.” The DCD Revision 17 via

TR-5, Revision 4 (Reference 1) provides the detailed requirements in those Information Items. 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. This change via TR-5 provides specific interface criteria, including necessary Information Items for the COL applicant. The 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, test 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 information items are deemed necessary to bring the certification information into 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 Fig.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; thus it is acceptable except for one issue that requires further investigation. The radwaste building was evaluated for its potential collapse on the nuclear island, demonstrating that it would not impair the structural integrity of the nuclear island 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 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 auto 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 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). This concern is identified as **Open Item OI-SRP3.7.2-SEB1-02**.

With regard to missiles generated by external events (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 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 is **Open Item OI-SRP3.3.2-SEB1-01**.

The staff reviewed this item, including all possible types of missiles generated and the associated kinetic energies produced as a result of external events. Upon closure of OI-SRP3.3.2-SEB1-01, the evaluation shows 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 roof rise 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 development of COL Information Items 3.3-1 and 3.5-1 committed in DCD Revision 15.

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 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 nuclear island safety-related structures.

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, pending resolution of the Open Items OI-SRP3.7.2-SEB1-02 and OI-SRP3.3.2-SEB1-01 as discussed above, 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.

The design of the annex, radwaste, and diesel/generator roofs now incorporates parapets with weir openings to drain water from the roof. The applicant, however, has not provided an analysis to show that these openings are sufficient to prevent ponding of water on the roof given the increase in the Probable Maximum Precipitation value from 19.4 in/hr to 20.7 in/hr (addressed in Chapter 2 of this SER) and the additional design features. The analysis should address the potential for blockage of the weir openings from ice and/or other debris. This concern is identified as **Open Item OI-SRP3.4.1-RHEB-01**.

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. The applicant, however, has not provided an analysis of the external flooding caused by tank rupture of the new tank design on safety-related structures, systems and components. This concern is identified as **Open Item OI-SRP3.4.1-RHEB-02**.

The staff 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 proposed changes to the AP1000 DCD also specify that no access openings or tunnels penetrating the exterior walls of the nuclear island are below grade and that a waterproof membrane or waterproofing system will be installed for the seismic Category I structures below grade.

The staff concludes that these proposed changes do not significantly impact the existing FSER Section 3.4.1.1.1 assumptions, findings or conclusions related to external flooding events or protection.

3.4.1.1.2 Conclusion

The staff reviewed the applicant's proposed changes to 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 in 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. Pending resolution of the open items described above, the staff finds that all of the changes to the AP1000 external flooding are acceptable because they are in compliance with 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.

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

m^3 (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 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" (pages 3.4-20-21):

  (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*** ~~800,000~~ gallon passive containment cooling system water storage tank above the valve room."
  (c) "Leakage will flow down to the landing at elevation ***277' 2″*** ~~264' 6"~~ 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 basis for these changes is described in TR-105.

- 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 TR 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 in 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 level’s 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 which 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 finds that the change does not affect the staff's conclusion documented in NUREG-1793 regarding this area of the auxiliary building. This conclusion is based on the following considerations: 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 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 nonsafety-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 ***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 nonsafety-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 significantly impact the existing SER Section 3.4.1.2 assumptions, findings, or conclusions related to internal flooding.

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

In its previous evaluations of the AP1000 DCD, Section 3.4.1, documented in NUREG-1793, 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, as documented in AP1000 DCD, Revision 15, 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 <u>Summary of Technical Information</u>

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 <u>Evaluation</u>

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 nuclear island 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 October 2, 2008 (ADAMS Accession Number ML082800327). The applicant stated that the increase in flood level would be 6 inches (15 cm) more, in addition to the flood level imposed by 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 open item.

This concern is identified as **Open Item OI-SRP3.4.2-SEB1-01**. Pending the resolution of this open item, the staff concludes that the change does not significantly impact the existing SER Subsection 3.4.1.1 assumptions, findings or conclusions related to external flooding events or protection based on 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." RAI-SRP2.4-RHEB-01 was presented to Westinghouse to clarify 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**. After receiving an acceptable response to the RAI and this open item is closed, the staff will have 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. Pending resolution of the open items described above, 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

3.5.3 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”), on site interface criteria for missiles generations and wind and tornado loadings by the Combined License applicant is fulfilled 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 Fig. 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 combined license 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 Fig. 1.2-2 of 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. 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 nuclear island structures or result in reclassification of their seismic categories.

The staff reviewed the analysis and found 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 nuclear island 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 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. This concern is identified as **Open Item OI-SRP3.7.2-SEB1-02**. Additional information on this Open Item is contained in Subsection 3.3.4 herein.

(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 cause a producing missile to 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, 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 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. Thus, the reanalysis of the shield building for a tornado-driven siding missile is **Open Item OI-SRP3.3.2-SEB1-01**.

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, pending the resolution of **Open Items OI-SRP3.7.2-SEB1-02** and **OI-SRP3.3.2-SEB1-01**, the staff finds that they are acceptable because they are in compliance with 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.

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

In the AP1000 DCD, Revision 17, the applicant 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 has proposed to revise those secondary, non-safety-related components that are used to mitigate postulated line ruptures. The applicant characterized this change as an editorial change that provides consistency with TR-86, "Alternate Steam and Power Conversion Design," (APP-GW-GLN-018), Revision 1, June 18, 2007.

(2) In DCD Section 3.6.1.3.3, "Special Protection Considerations," the applicant has proposed to delete the following statement in the criterion requiring protection 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 would instead state: "Instrumentation required to function following a pipe rupture is protected." 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, "Specific Protection Considerations," the applicant provided COL actions that reference back to the design basis criteria in DCD Section 3.6.1. In addition, the applicant has now proposed to further revise these COL actions based on the information provided in TR-92, "AP1000 Optimized Condenser Design," (APP-GW-

GLR-021), June 30, 2007 and TR-7, “Pipe Break Hazards Analysis,” (APP-GW-GLR-074), January 2007. The staff’s evaluation of this change is discussed in Section 3.6.4, “Combined License Information,” of this SER.

3.6.1.2 Evaluation

The staff reviewed all changes to Section 3.6.1 in the AP1000 DCD Revisions 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’s review of DCD Section 3.6.1 was restricted 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.

3.6.1.2.1 Design Basis Assumptions

In DCD Revision 17, 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-energy systems 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 the DCD Section 3.6.1, the staff identified areas in which additional information was necessary to complete its evaluation of the applicant’s change. In DCD Section 3.6.1.1 to paragraph J, the applicant amended this 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 the DCD, Section 3.6.1.3.3, “Specific Protection Considerations,” the secondary components list consists of the turbine stop, the moisture separator reheater stop, and the turbine bypass valves. This is 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 (Agencywide Document Access and Management System (ADAMS) Accession Number ML081900157), 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 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), “The 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 its 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, “Safety Design Basis,” 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.

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. The staff confirmed that the markup has been incorporated into the AP1000 DCD, Revision 17, Section 3.6.1.1, Paragraph J; Section 3.6.1.3.3; and Section 10.3.1.1.

3.6.1.2.2 Protection Mechanisms

In DCD Revision 17, 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 is an inconsistency between 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, May 2007 and the DCD revision that needs 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 the technical report, the applicant deleted the entire second bullet, while in Revision 16 to the DCD, the first sentence of the second bullet remains (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 (Agencywide Document Access and Management System Accession Number ML081900157), the applicant stated that in developing the markup for the 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 second bullet to DCD Section 3.6.1.3.3, Revision 16 is accurate with respect to the design and applies to all safety-related instrumentation in a harsh environment. 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 Information Items

In DCD Revision 15, the applicant had included COL Information Item 3.6-1, which instructed the COL applicant to complete the pipe break hazard analysis. DCD Revision 15 Sections 3.6.1 and 3.6.2 provided all the design criteria that the COL information item would be demonstrating. In DCD Revisions 16 and 17 the applicant proposed to eliminate this COL information item. In order to support the removal of this COL information item from the DCD, the applicant provided a pipe break hazard analysis report. The staff determined that this report was incomplete and did not address all the information that the COL Information Item 3.6-1 specified. The complete staff evaluation of this proposed change is addressed in Section 3.6.2 of this SER.

As described in Section 3.6.2 of this SER, the applicant responded to RAI-SRP3.6.2-EMB2-01, in letters dated June 20, 2008 (ADAMS Accession Number ML081780176), August 15, 2008 (ADAMS Accession Number ML082330096), December 5, 2008 (ADAMS Accession Number ML083440071), June 30, 2009 (ADAMS Accession Number ML091870126 and ML091870127) and July 22, 2009 (ADAMS Accession Number ML092050157). In its latest response, the applicant stated that the pipe break hazard analysis report will be completed and available for the staff's review by December 31, 2009. The staff cannot determine that the piping design in the AP1000 meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena"; and GDC 4, "Environmental and Dynamic Effects Design Bases," until the pipe break hazard analysis report is completed. Therefore, the staff concerns related to the proposed deletion of COL Information Item 3.6-1, "Pipe Break Hazard Analysis," is still unresolved. This concern is identified as **Open Item OI-SRP3.6.2-EMB2-01**.

3.6.1.3 Conclusions

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's meeting its relevant requirements in 10 CFR Part 50, Appendix A, GDC 2 and in GDC 4. 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 finds that, pending resolution of Open Item OI-SRP3.6.2-EMB2-01 and pending staff acceptance of the proposed changes to DCD Section 3.6.2, 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, Revisions 17. On the basis that the AP1000 postulated piping failures in fluid systems outside containment continue to meet all applicable acceptance criteria, the staff finds that all of the changes to 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

In the AP1000 DCD, Revision 17 the applicant proposed changes to Subsection 3.6.4.1,"Pipe Break Analysis." This subsection identifies a COL Information Item 3.6-1, "Pipe Break Hazards Analysis" which states:

> 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 subsections 3.6.1.3.2, "Protection Mechanisms" and 3.6.2.5, "Evaluation of Dynamic Effects of Pipe Ruptures." The as-built pipe rupture hazards analysis will be documented in an as-built Pipe Rupture Hazards Analysis Report.

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 17, APP-GW-GLR-021, APP-GW-GLR-074, and the applicant's letters dated January 14, 2008 (ADAMS Accession Number ML080160253), and December 5, 2008 (ADAMS Accession Number ML083440071). In APP-GW-GLR-021 and APP-GW-GLR-074, the applicant proposed to modify the COL Information Item and provided a pipe break hazards analysis report for the staff's review. The applicant stated that the report addressed and documented, on a generic basis, design activities required to complete the COL Information Item in Section 3.6.4.1 in the AP1000 DCD. The applicant further stated that when the staff's 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 AP1000 Design Certification. On the basis of the review of the 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 break analysis documented in APP-GW-GLR-074 cannot 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, the applicant proposed to revise Subsection 3.6.4.1 of the DCD to address the staff's comments on the completeness of APP-GW-GLR-074. Based on the staff's review of the information included in DCD, the staff determined that the following additional information concerning the acceptability of the proposed COL holder item was needed:

1a. The staff maintains that the pipe break hazards analysis report of 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 to the COL Information Item that cannot be resolved completely before the issuance of a license. It requires the applicant to provide sufficient information to justify why that item cannot 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 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 I tables of System Based Design Description and ITAAC, the applicant includes an acceptance criterion that states that for the as-built piping, a pipe break evaluation report exists and concludes that protection from the dynamic effects of a line break is provided. It should be noted that the pipe break hazards analysis report is required for all the piping systems (with the exception of LBB piping) that are within the scope of SRP 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 break analysis performed in accordance with the criteria outlined in subsection 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 break analysis and is not part of the as-built reconciliation. Westinghouse is requested to address this concern and to revise the DCD 3.6.2.5 accordingly.

The requests for additional information are documented in RAI-SRP3.6.4-EMB2-01 for question 1a and 1b (ADAMS Accession Number ML081230062) and RAI-SRP3.6.2-EMB2-01 for Question 2 (ADAMS Accession Number ML081650249).

By letter dated December 5, 2008 (ADAMS Accession Number ML083440071), Westinghouse provided its response to the above RAIs. Based on the staff's review of the applicant's response, the staff concluded that the as-built reconciliation of the pipe break hazards analysis report as included in the ITAAC tables of the DCD was previously reviewed and found acceptable by the staff. However, with respect to the as-designed pipe break hazards analysis, the staff found that the applicant had not yet adequately addressed the staff's concern relating to the completion of the as-designed Piping Hazards Analysis Report issue. Specifically, it is not clear that the as-designed pipe break hazards analysis report will include all piping systems within the scope of SRP 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. Westinghouse's RAI response did not clearly address the process including the milestone for the completion of the as-designed pipe break hazard analysis reports for all piping systems within the scope of SRP 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 the staff to discuss its plan, schedule and scope of the as-designed pipe break hazard evaluations. The meeting was held on May 20, 2009 at Westinghouse Twinbrook office in Rockville, Maryland. During the meeting, Westinghouse indicated that it will complete an as-designed pipe break hazard evaluation 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 SRPs

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 will be completed by COL applicants referencing the AP1000 certified design. In addition, Westinghouse indicated that it will include all the above information in an RAI response to address the staff's concerns related to the as-designed piping break 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 this as-designed pipe break hazards analysis report. Also, Westinghouse is 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 representatives of the ones that will be used in AP 1000 design.

By letters dated June 30 and July 22, 2009 (ADAMS Accession Numbers ML091870127 and ML092050157, respective) 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 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 break hazards analysis report, with the exception of some pipe whip restraint and jet shield designs, is to be completed by December 31, 2009 and that some pipe whip restraint and jet shield designs are not expected to be completed in time to support the 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 may be completed during the COL application review or after the license is issued. It should be noted that during the May 20, 2009, meeting, Westinghouse indicated that to support the staff's audit, it will complete an as-designed pipe break hazards evaluation 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 non-safety-related piping systems that were not addressed in Westinghouse RAI responses) within the scope of SRPs 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 is not clear as to what pipe whip restraints and jet shields design will be completed by December 31, 2009, and how they are representative of the ones that will be used in 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 break 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 appears that the final completion of all pipe whip restraint and jet shield design is a COL Information Item, however, it is not clearly labeled as one. Westinghouse is requested to clearly identify it as a COL Information Item or to make it an ITAAC item.

Pending a satisfactory resolution of the above-described staff's concerns and a satisfactory audit of the as-designed pipe break hazards analysis report, the as-designed pipe break hazards analysis report for AP1000 constitutes **Open Item OI-SRP3.6.2-EMB2-01.**

3.6.2.3 Conclusion

Pending the satisfactory resolution of **OI-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 <u>Evaluation</u>

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 its 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 is based on the piping stress analysis results using seismic loadings associated with an AP1000 plant situated on a hard-rock site. At this time, Westinghouse is 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 will be completed in conjunction with piping DAC. The NRC staff will review as-designed LBB analyses results before the design certification amendment is issued as part of its resolution of AP1000 piping DAC to verify that the bounding analysis curves for piping for the other-than-hard-rock seismic input remains bounding. The concern is identified as **Open Item OI-SRP3.6.3-CIB1-001**.

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 Design Control Document (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 needs to be performed by the staff in accordance with the ITAAC scope as clarified above.

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 leak-before-break (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 <u>Conclusion</u>

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 upon resolution of **Open Item OI-SRP3.6.3-CIB1-01** and upon acceptable resolution of **Open Item OI-SRP3.6.3-CIB1-01**, COL Information Item 3.6-2 may be closed.

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.

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

Later

3.8 Design of Category I Structures

Later

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, in its response to RAI-SRP3.9.1-EMB1-03 the applicant added WESTEMS design computer code to AP1000 DCD Table 3.9-15 for application of the fatigue analysis of components. The WESTEMS computer program was not previously reviewed and approved by the NRC.

3.9.1.1 Technical Evaluation

Field Code Changed

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 that the 2,000 occurrences were split 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 it has occurred 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 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 conservative occurrences 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 on-site 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 March 5, 2008, response to the staff's RAI-SRP3.9.1-EMB1-03 Revision 2, 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. However, Westinghouse failed to provide the staff with evidence of the computer code verification and validation documentation for design of the ASME Class 1, 2 and 3 components and piping in accordance with Appendix B to 10CFR 50.55a or ASME Code NQA-1. Instead, it stated that the WESTEMS documentation package will be made available for additional NRC review. On May 26 - 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. A follow-up audit will be performed in the Westinghouse Twinbrook Office in Rockville, Maryland to allow review of the remaining documents as they relate to WESTEMS. **This concern is identified as Open Item OI-SRP3.9.1-EMB1-03**.

During the audit, the NRC discussed with the applicant the theoretical background, formulation, validation methods, and benchmarking problems pertaining to WESTEMS.

The transfer function stress database input of WESTEMS program was developed by applying unit temperature step increase with a specific temperature's material property to the component model. However, the design/operating transients temperatures may vary significantly. The staff noted that transfer function stress database has to be properly benchmarked to avoid stress result deviation due to inadequate temperature selection for every component problem to be used in WESTEMS transfer function method. The staff requested that the applicant provide and document guideline/criteria for developing/benchmarking transfer function stress database. This concern is identified as **Open Item OI-SRP3.9.1-EMB1-04**.

The staff reviewed the basis documents for WESTEMS during the on-site review. In CN-PAFM-06-159, “WESTEMS Software Change Specification for Version 4.5,” the applicant generated an algebraic stress histories option to be used in selection of peak and valley times. The option used equations to calculate time vs. stress in selecting peak and valley times.

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 is identified as **Open Item OI-SRP3.9.1-EMB1-05**.

The staff reviewed WESTEMS validation package CN-PAFM-06-161. The applicant’s validation package compared WESTEMS results with results of MAXTRAN79 and THERST. The applicant stated that the comparison used slightly different material properties. The comparison also showed the results are different with different programs. However, the applicant considered that the validation was acceptable even with a significant difference in ΔT calculation and stress result comparison. The staff noted that computer program benchmark must use the same input model in alternate calculations or hand calculations. The staff noted that use of a slightly different model and different material properties to compare the results with approximation may not be adequate to benchmark a computer program. The staff requested the applicant to provide benchmark acceptance criteria to validate the computer code calculation. This concern is identified as **Open Item OI-SRP3.9.1-EMB1-06**.

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 is identified as **Open Item OI-SRP3.9.1-EMB1-07**.

3.9.1.2 Conclusions

Based on the information provided in Westinghouse’s responses to the RAIs, the staff finds that the applicant did not provide sufficient information regarding the qualification of the WESTEMS computer code. The AP1000 DCD application will not meet the guidance provided in Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants,” June 2007, until the open items identified in the previous section are satisfactorily resolved.

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 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 internals 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 (in-service) 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 inch. 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 diminished likelihood of large vortex formation in the lower plenum region. However, until the scale model flow test, including the flow skirt and its connections to the reactor vessel, is complete, and the staff reviews the test results, this remains **Opem Item OI-SRP3.9.2-EMB1-07.**~~

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.

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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 feet). 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. Therefore, the staff contends that the above 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 a COL Action Item, 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 to be 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 Technical Reports.

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 stress corrosion cracking (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 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 ºC (608 ºF) 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 ºC (608 ºF) 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 ºC (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 that Westinghouse provide 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 that Westinghouse 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 they applied the screening criteria for determining susceptibility to IASCC.

In RAI 2, the staff requested that Westinghouse 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 that Westinghouse 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 that Westinghouse 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 that Westinghouse 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 that Westinghouse 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 that Westinghouse 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 ºC (608℉) 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 that Westinghouse 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 meets 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 ºC (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 ºC (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 ºC (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 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 that further examinations of reactor internal components for void swelling behavior is 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 that Westinghouse 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 that the IASCC neutron fluence threshold from a previous Westinghouse report, WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals," is 1 x 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 1-2 x 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 x 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 that 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 that Westinghouse 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 x 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 that 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 x 10^{21} n/cm^2 (E > 1.0 MeV), Westinghouse stated that the AP1000 reactor internal components were screened using the MRP-175 screening recommendations without regard to the components' thermo-mechanical history. Westinghouse further stated that it is not anticipated that a material's degree of cold work will necessitate screening criteria that are different from the criteria recommended by MRP-175. The staff found that this response adequately resolved RAI 11 because Westinghouse adequately explained why the MRP-175 IASCC screening recommendations were applied irrespective of the components thermo-mechanical history. 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 that Westinghouse 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 that 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 that 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 that 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 that IASCC studies reviewed in MRP-175 have shown that the IASCC resistance of nickel-based alloy X-750 is approximately the same as that for Type 304 and 316 austenitic SSs. Furthermore, AP1000 reactor internal reactor internal components that are 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 that this response adequately resolved RAI 13 because Westinghouse adequately addressed IASCC screening of reactor internal components fabricated from nickel-based alloys.

Studies have shown that 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 that 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 heatup or cooldown 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 that 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 that a potentially important aspect of void swelling in PWRs arises from transmutation of trace amounts boron, preexisting in most austenitic SSs, to produce lithium and helium. Section G.1 of MRP-175 indicates that 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 that Westinghouse 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 that 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 that this response adequately resolved RAI 15 because Westinghouse explained that the MRP-175 void swelling screening recommendations were applied irrespective of the components' helium content. The staff justified 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 that 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 within the nuclear power industry and the open literature that identify 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 that 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 that the IASCC screening criteria were appropriate for evaluating stainless steel reactor internal components to determine their susceptibility to IASCC behavior.

Notwithstanding the limitations toward establishing all-encompassing IASCC criteria for reactor internal components in PWRs, the staff found that 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 for reactor internal components that 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 that 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 that 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 that 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 that 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.

3.9.2.4.5 Conclusions

The staff finds that the evaluation of the AP1000 reactor internal components for IASCC and void swelling meets the requirements of ASME Section III based on the MRP-175 screening criterion as reported in TR-12 and is acceptable.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Introduction

DCD Tier 2, Subsection 3.9.8.2, addresses Combined License Information Item 3.9-2 for the design specifications and reports for the major ASME Code, Section III components. DCD Tier 2, Subsection 3.9.8.2 states that the design specifications and design reports for the major ASME Code, Section III components are available for NRC audit via the technical reports listed in Table 3.9-19. Subsection 3.9.8.2 also states that design specifications and selected design analysis information are also available for ASME Code, Section III valves and auxiliary components. Westinghouse has proposed that these requirements be deleted from the DCD.

3.9.3.2 Evaluation

The Westinghouse letter dated February 8, 2008 (ADAMS Accession Number ML080440066) states that 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-SRP3.9.3-EMB2-01, the staff requested Westinghouse to verify that the schedule provided in the letter is still valid. If not and there is a chance that the design reports and specifications will not be available prior to completion of the review, Westinghouse is requested to provide a justification for changing the COL Item from an applicant to a holder item. If the component design is to be completed after the COL issuance, the staff requested Westinghouse to describe an established procedure to allow addressing the COL item through a design ITAAC. In order to ensure a timely closure of the COL information item, the staff requested Westinghouse to commit to completing the design of the above mechanical components, and make the associated design specifications and design reports available for audit prior to their fabrication and installation.

By letter dated June 26, 2008 (ADAMS Accession Number ML081820723), Westinghouse stated that review of design specifications and as-designed design reports for ASME Code Section III components provides a means for the NRC to verify that 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 that require as-built design reports for the ASME Code, Section III components. Westinghouse stated that 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 that 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 that 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 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 that the revision of the COL Information Item in DCD Revision 16, Subsection 3.9.8.2, was based on the expectation that 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 that 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 that it is expected that 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 that was restated as a COL holder item is related to the as-built reconciliation of thermal cycling and stratification loadings on piping.

Westinghouse stated that it has completed the design specifications for the major ASME Code, Section III components in the AP1000. These are available for review by the NRC. These specifications include a Professional Engineer certification. The as-designed design reports and supporting analysis for most of the major components are also complete and available for review. Westinghouse stated that the analyses include the use of the updated (six soils case) seismic design spectra. The components that have or will have an as-designed design report ready for review on a schedule to support the NRC preparation of the SER with open items include: reactor vessel, control rod drive mechanism, steam generator, pressurizer, passive RHR heat exchanger, core makeup tank, accumulator, and reactor internals. Westinghouse stated that a reactor coolant pump design report will be available for review later and may require an open item in the SER. Westinghouse stated, however, that it will be available for review well before open items need to be cleared for the advance SER. Westinghouse stated that 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 open items 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 will be able to conclude whether the COL information item is closed.

In its letter of June 26, 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 technical reports 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-SRP3.9.3-EMB2-01. Subsequently, the staff reviewed AP1000 DCD, Revision 17, when it became available, and verified that the latest revision of the DCD has incorporated all the changes as required.

During October 13 - 17, 2008, the staff conducted an on-site 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 that 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 that 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-SRP3.9.3-EMB2-05 that Westinghouse demonstrate how the Westinghouse methodology meets the ASME Code for J-groove weld design.

In response to RAI-SRP3.9.3-EMB2-05, Westinghouse stated that it has satisfied the intent of Paragraph NB-3228.5 (a) of the ASME Subsection NB. According to the 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 and, to a lesser degree, the axial stress. The ratcheting mechanism cannot occur from 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. 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 July 15, 2009, response to RAI-SRP3.9.3-EMB2-05 Revision 2 (ADAMS Accession Number ML091980041) 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. This conservative approach does not satisfy the limits of NB-3228.5(a), so 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, 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 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.

The staff reviewed the Westinghouse response and found it unacceptable. Westinghouse in its response stated that the design does meet the intent of the code. This conclusion is based on Westinghouse's interpretation of the code intent. Westinghouse did not provide reference to a past precedent or an approved code case which supports Westinghouse interpretation. The staff cannot accept Westinghouse interpretation without an approved code case or past precedent. This concern is identified as **Open Item OI-SRP-3.9.3-EMB2-05**.

The staff reviewed the design specification and other supporting documents associated with Containment Recirculation Screens and found several issues that are incompletely addressed in the design specification. The staff requested in RAI-SRP3.9.3-EMB2-08 that Westinghouse 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) The loading conditions and combinations are incompletely presented in the documents reviewed by the staff. Provide the following: (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) While it is possible to design containment cleanliness programs to sustain low latent debris inventory in containment, justify the latent debris mass value used for the screen pressure drop component of the structural load on the IRWST and sump screens. Additionally, justify that the flow rate through the screen is conservatively calculated.

This concern is identified as Open Item OI-SRP-3.9.3-EMB2-08.

Based on the audit, the staff has reached the conclusion that, with the exception of the two open items, the AP1000 component design has been completed to an extent such that the COL Information Item in Revision 15 is met, allowing the aspects of the COL Information Item addressing components to be eliminated.

Since its original Design Certification, 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 26, 2008, Westinghouse stated that Westinghouse has changed the design basis for the major ASME Code, Section III components to include the design spectra and seismic requirements that envelope 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 that the analyses supporting the as-designed design reports prepared or being completed for NRC review were in compliance with the design specifications that include these revised seismic requirements of the six soils case. Based on the above response and the confirmation obtained from the staff's on-site review, the staff found that Westinghouse has adequately incorporated the latest revised seismic input motions for the component design. RAI-SRP3.9.3-EMB2-02 is, therefore, closed.

The staff's review of the 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.

Piping-related issues are discussed in Section 3.12 of this SER.

3.9.3.3 Conclusions

Based on the information provided in the Westinghouse responses to the RAIs, the staff concludes that, pending resolution of the open items identified by the staff during the audit, the proposed changes addressed herein described in DCD, Revision 17, meet the requirements of 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress limits specified for ASME Code Class 1, 2, and 3 components by ensuring that systems and components are designed to quality standards commensurate with their importance to safety, and that these systems can accommodate the effects of such postulated events as LOCAs and the dynamic effects resulting from earthquakes. The specified design and service combinations of loadings, as applied to ASME Code Class 1, 2, and 3 pressure-retaining components in systems designed to meet seismic Category I standards, provide assurance that, in the event of an earthquake affecting the site or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress limits for the materials of construction. Limiting the stresses under such loading combinations provides an acceptable basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

3.9.4 Control Rod Drive Systems

In Revision 17 to the AP1000 DCD, Westinghouse Electric Company, LLC (Westinghouse) proposed changes to the hydrostatic test pressure for the control rod drive mechanism (CRDM) housing as well as other materials related 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.

In a future DCD revision, DCD Tier 2 Table 3.2-3 will be modified 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 following an earthquake requiring plant shutdown.~~

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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 that 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 that the attachment of the latch assembly housing to the vessel head is accomplished by a shrink-fit and partial penetration weld. Westinghouse determined that 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 that the proposed changes are an editorial change to the AP1000 DCD and as such does not affect the design basis of the component. Furthermore, the proposed change describes the correct fabrication sequence, uses the correct terminology and it is, therefore, 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 that 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 that a reactor trip occurs when power to the stationary as well as 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

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The staff became aware of discussions internationally concerning the classification and qualification of the CRDM latch assembly. Based on these discussions, the staff determined that the adequacy of seismic qualification of the Control Rod Drive Mechanism (CRDM) for AP1000 standard design may not be adequate. GDC 2 of Appendix A to 10 CFR Part 50 states that 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. As AP1000 is a standard design, the staff requested clarification.

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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.

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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 precedence of operating plants, and 3) the function of the latch assembly and coil stack assembly in the AP1000 CRDM.

The staff finds Westinghouse’s justifications using the precedence 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 as justification for not seismically qualifying the latch assembly. 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. However, for justification 1 (design finality), the staff, in NUREG-1793 reviewed and certified Rev. 14 of the AP1000 DCD. Since then, the CRDM design planned for certification in the United States has not changed. 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 U.S. NRC requirements, they expect that 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 are performed to demonstrate compliance with the requirements of 10 CFR 50.62. These analyses demonstrate that 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 that 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 that 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. However, the proposed DCD revision has not been incorporated into AP1000 DCD Revision 17. It should be incorporated into the future DCD revision. RAI SRP3.9.4-EMB1-01 will remain a **Confirmatory Item**.

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3.9.4.2 Conclusion

The staff further concludes that 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 that the changes to the CRDM design description provided in AP1000 DCD, Revision 17, are acceptable. The staff finds that the conclusion of NUREG-1793, specifically that the design of the CRDS for the AP1000 meets GDC 1, 2, 14, 29 and 10 CFR 50.55a, remains valid.

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 that 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 that 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 that the preload is maintained and that design limits are achieved. The applicant also stated that 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 that 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 that the attachment weld between the flow skirt and the reactor vessel flow skirt support lug is 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 that 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 that 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 received in a letter dated June 20, 2008, the applicant stated that the primary function of the flow skirt is to assure that 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 Computerized 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 received in a letter dated June 20, 2008, the applicant provided a figure of the flow skirt which clarified its form and function, and committed to including this figure in Revision 17 of the AP1000 DCD.

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 that 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 that 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 that 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~~established the design requirements according to the provisions of ASME III, Subsection NG. Therefore, the staff concluded that 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 that the results of the Trojan 1 reactor tests showed that 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 that there is reasonable assurance that 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 that there are no adverse effects of FIV and flow-excited acoustic resonances on the reactor vessel internal structures. On this basis, the staff finds that 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 <u>Conclusion</u>

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 that 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 that acceptable fuel design and performance limits are met during conditions of normal operation and anticipated operational occurrences.

The staff concludes that 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 <u>Evaluation</u>

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 safety-related 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 that 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 that 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 that a COL ~~A~~applicant referencing the AP1000 design needs to fully describe the IST, ~~m~~Motor-~~o~~Operated ~~v~~Valve (MOV) ~~t~~Testing and other operational programs as defined in Commission Paper SECY-05-197, “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 ~~A~~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 Final Safety Analysis Report (FSAR) in a COL application to support a reasonable assurance finding of acceptability. A COL ~~A~~applicant may rely on information in the applicable Design Certification ~~DCD~~ 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 that the AP1000 DCD will address issues common to COL ~~A~~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 ~~A~~applicant in fully describing the operational programs in support of its COL application for an AP1000 reactor. As described below, the NRC staff concludes that the revision to Subsection 3.9.6 of the AP1000 DCD continues to provide an acceptable description of the functional design, qualification, and IST p~~P~~rograms for pumps, valves, and dynamic restraints sufficient for ~~in~~ 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 open and confirmatory items in this section.

A COL Applicant may reference the provisions in Subsection 3.9.6 the AP1000 DCD as part of its responsibility to fully describe the IST, MOV T~~t~~esting, and other operational programs in support of its COL application.

AP1000 DCD Tier 2, Section 3.9.6 states that Table 3.9-16~~.~~, “Valve Inservice Test Requirements,” identifies the components subject to the preservice and IST programs, and the method and frequency of preservice and inservice testing. The NRC staff will evaluate the full description of the IST Program provided by a COL a~~A~~pplicant 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 P~~p~~rogram 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 that 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 that 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 that 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 that 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, PA. The staff found that 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 open items (ADAMS Accession Number ML083110154). The audit response was tracked as **Open Item 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 that a reference to ASME QME-1-2007 will be included in AP1000 DCD Tier 2, Section 3.9. Second, Westinghouse stated that 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 that 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 that the applicable valve design specification indicates that active valves must be qualified in accordance with the ASME QME-1 standard, and that the specification will be further clarified to indicate that any existing testing used to demonstrate functional qualification must fully satisfy the provisions of ASME QME-1-2007. Fourth, Westinghouse stated that 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. The NRC staff considers Westinghouse to have provided an acceptable plan to resolve the audit follow-up items. **Open Item OI-SRP3.9.6-CIB1-01** will remain open pending a follow-up audit to review the AP1000 design specification changes. **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.~~On October 14 and 15, 2008, the NRC staff conducted an onsite review of design and procurement specifications for pumps, valves, and dynamic restraints to be used for the AP1000 reactor at the Westinghouse offices in Monroeville, PA. The staff found that 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. The NRC is preparing a revision to RG 1.100 to address the acceptability of ASME QME-1-2007. The staff found that the AP1000 valve design specifications did not include a specific calculation method accepted by Westinghouse, or the preferred approach for consideration of various parameters and their bias error and random uncertainty, in the determination of actuator output capability and valve operating requirements. Westinghouse indicated that a specific calculation method had not been included in the design specification in order to allow flexibility for the vendor in its proposed approach, but that the vendor analyses would be evaluated by Westinghouse prior to acceptance. At the conclusion of the onsite review, the staff discussed its findings from the review of AP1000 design and procurement specifications. Westinghouse indicated that the staff comments would be addressed in a future revision of the specifications. The staff considered the following items to remain open from the onsite review: (1) the absence of a reference to ASME QME-1-2007 in AP1000 DCD Tier 2, Section 3.9, (2) the need to provide a basis for the seat coefficient of friction assumptions for gate and globe valves, (3) the need to clarify that vendors must satisfy the QME-1-2007 qualification requirements in addition~~

~~to the specific testing indicated in the design specifications, and (4) the need to resolve the difference between the RAI response for check valve testing and the piping diagram for check valves PXS-V016A/B and V017A/B. In a memorandum dated November 6, 2008, the NRC staff documented the results of the onsite review with the specific open items (ADAMS Accession Number ML083110154). Westinghouse needs to resolve the open items from the onsite review to close RAI-SRP3.9.6-CIB1-01. This concern is identified as **Open Item OI-SRP3.9.6-CIB1-01**.~~

AP1000 DCD Tier 2, Section 3.9.2, "Dynamic Testing and Analysis," describes tests to confirm that piping, components, restraints, and supports have been designed to withstand the dynamic effects of steady-state flow-induced vibration (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 that the purpose of the expansion, vibration and dynamic effects testing is to verify that 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 clarify the use of nonintrusive techniques within the IST P~~p~~rogram 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 ML061280315), Westinghouse stated that 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 that inservice testing may incorporate the use of nonintrusive techniques to periodically assess degradation and performance of selected check valves. The NRC staff finds that 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 that the COL ~~holder~~ licensee will define any nonintrusive techniques that 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 that 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 Joint Owners Group (JOG) Program for MOV Periodic Verification. 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 P~~p~~rogram 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 that 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 that 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 that lessons learned from GL 96-05 and the JOG MOV P~~p~~eriodic V~~v~~erification ~~study~~ Program are reflected in the IST Program and valve procurement testing requirements. Revision 17 to the AP1000 DCD indicates that 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 that 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. RAI-SRP3.9.6-CIB1-03 is closed.

AP1000 DCD Tier 2, Subsection 3.9.6.2 states that 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 that 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 that 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 that 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 that 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 a~~A~~pplicant 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 that 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 that the MOVs are capable of performing their design-basis safety functions. In RAI-SRP3.9.6-CIB1-08, the NRC staff requested that Westinghouse 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 that 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 Open Items OI-SER3.9.6-CIB1-02, 03, 04, and 05.~~As a result, Revision 17 to AP1000 DCD Tier 2, Subsection 3.9.6.2.2, specifies that safety-related POVs are required by the procurement specifications to have the capability to perform diagnostic testing to verify that the valves can perform their design-basis safety functions. For POVs that meet the JOG MOV Program requirements, the initial test frequency will be consistent with the JOG MOV Program based on the valve risk ranking and margin. POVs meeting the JOG MOV Program will be statically tested consistent with the JOG MOV Program with a maximum test frequency of once every 10 years. For POVs that do not meet the JOG MOV Program, the initial test frequency will be based on the functional margin determined from the ASME Standard QME-1 and baseline testing with supplementary analysis covering uncertainties and risk ranking. The initial test frequency will be in accordance with ASME OM Code Case OMN-1, paragraph 3.3.1, until sufficient data are collected. POVs that do not meet the JOG MOV Program will have a combination of static and dynamic tests performed to confirm operability and develop the basis for future testing. POVs that meet the JOG MOV Program will use the methodology in the JOG MOV Program for functional margin. For POVs that do not meet the JOG MOV Program, the functional margin will be determined by analysis and supplemented by QME-1 testing with uncertainties taken into account. Valves for which functional margins have not been determined will require dynamic testing to determine appropriate margins. Also in response to this RAI, Revision 17 to the AP1000 DCD includes changes to the Technical Specifications and Technical Specification Bases to correct the reference for AP1000 IST Program activities from the ASME BPV Code, Section XI to the ASME OM Code.~~

As Open Item OI-SRP3.9.6-CIB1-02, the NRC staff tracked the need for ~~Several items need to be addressed to resolve this RAI. First,~~ the reference to static testing of valves in the AP1000 DCD ~~needs~~ 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 that 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. During discussions of this planned DCD change, Westinghouse indicated that the valve design specifications will require that safety-related MOVs to be used at AP1000 plants be within the scope of the JOG MOV Periodic Verification Program. The staff considers that these planned DCD changes will resolve this portion of RAI-SRP3.9.6-CIB1-08. The NRC staff will confirm the incorporation of a provision in the AP1000 valve design specifications that MOVs be within the scope of the JOG MOV Periodic Verification Program during a follow-up audit. 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**. The review of the valve design specifications for the provision that MOVs must be within the JOG MOV Periodic Verification Program scope will be performed as part of **Open Item OI-SRP-3.9.6-CIB1-01**.

As Open Item OI-SRP3.9.6-CIB1-03, the NRC staff tracked the need for ~~(OI-SRP3.9.6-CIB1-02). Second,~~ the AP1000 DCD ~~needs~~ 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 that the AP1000 DCD Tier 2, Section 3.9 would be revised to reference ASME QME-1-2007. The staff considers that 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 Open Item OI-SRP3.9.6-CIB1-04, the NRC staff tracked the need for ~~(OI-SRP3.9.6-CIB1-03). Third,~~ 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 ~~as part of~~ the AP1000 IST Program ~~needs~~ to be ~~implemented~~ consistent with the edition of Code Case OMN-1 accepted in Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," or to indicate the need for the submission of ~~or~~ 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 that 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 that this planned change to the AP1000 DCD will resolve this portion of RAI-SRP3.9.6-CIB1-08. Therefore, OI-SRP3.9.6-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 Open Item OI-SRP3.9.6-CIB1-05, the NRC staff tracked the need for ~~must be submitted (OI-SRP3.9.6-CIB1-04). Fourth,~~ the Technical Specifications and Technical Specification Bases need 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 that 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**. ~~(OI-SRP3.9.6-CIV1-05). These concerns are identified as **Open Items OI-SRP3.9.6-CIB1-02, 03, 04, and 05**, respectively.~~

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 that check valves must be exercised in the open and closed directions. In RAI-SRP3.9.6-CIB1-09, the NRC staff requested that Westinghouse 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 ~~stated that all AP1000 check valves can be full stroke exercised with flow without the need for nonintrusive techniques. In the future, Westinghouse stated that a licensee might use nonintrusive techniques in accordance with ASME OM Code, Subsection ISTC-5221, "Valve Obturator Movement." Westinghouse~~ specified that 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 noted that all check valves can be exercised to verify open and closed functionality, except as indicated in response to RAI-SRP3.9.6-CIB1-12.~~ Westinghouse stated that 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~~considers~~ the RAI response to be acceptable, but that the AP1000 DCD needed~~needs~~ to include the specified acceptance criteria for check valve testing. ~~Further, the reference in the RAI response to RAI-SRP3.9.6-~~

~~CIB1-12 needs to be clarified.~~ The staff tracked this item ~~This concern is identified~~ as **Open Item 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, OI-SRP3.9.6-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 AP1000 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 AP1000 IST Program for safety and relief valves. In response to this RAI in a letter dated September 9, 2008, Westinghouse stated that 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 AP1000 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 AP1000 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 that 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 that the applicable valves are subject to operability testing per the NRC regulations in 10 CFR Part 50, Section 50.55a. The NRC staff considered that Note 31 needed ~~However, Note 31 continues to specify a test frequency of the longest of every three refueling outages or 5 years until sufficient data exist to determine that a longer test frequency is appropriate in accordance with GL 96-05. The staff considers Note 31 to be inconsistent with the guidance in the JOG MOV periodic verification program, which provides a test frequency based on margin and risk ranking ranging from 2 to 10 years. Note 31 needs~~ to be clarified to be consistent with the JOG MOV P~~p~~eriodic V~~v~~erification P~~p~~rogram, and to ~~. Further, the AP1000 DCD should include the~~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 ~~This concern is identified~~ 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 that 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 that the JOG approach will be applied to all actuator types and that the attributes of the POV programs will include lessons learned as delineated in RIS 2000-03. The staff considers that 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 that 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 that 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 that 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 that 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 that 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 that 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 that passive valves with remote position indication will be locally observed to verify that 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 that 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 that 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 and that the fail safe~~ test is not identified as a separate 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 that 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 that remote position indication is used as applicable to verify proper fail safe operation, provided that 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 that ~~T~~the reference to ASME OM Code, Subsection ISTC-3700, need~~s~~ed to be clarified to confirm that the exercise test frequency requirements specified in the ASME OM Code for these valves will be satisfied. This item was tracked ~~concern is identified~~ as **Open Item OI-SRP3.9.6-CIB1-08.** In its letter dated January 26, 2010, Westinghouse noted that 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 that 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 AP 1000 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 that 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 that 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 that 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 that 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. RAI-SRP3.9.6-CIB1-16 is closed.

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 that 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 that 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 that 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 that AP1000 DCD Tier 2, Table 3.9-16 will be revised to specify that this test will be performed once every refueling cycle. The NRC staff finds that 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~~is~~ **Confirmatory Item CI-SRP3.9.6-CIB1-09~~1~~**.

Subsection 3.9.6.2.2 of the AP1000 DCD Tier 2 under Pressure/Vacuum Relief Devices states that 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 that 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 that the Westinghouse response to RAI-SRP3.9.6-CIB1-18 as incorporated into AP1000 DCD Revision 17 provides an acceptable clarification to ensure that 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 that 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 that 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 chemical volume and control system (CVS) containment isolation valves CVS-PL-V045, CVS-PL-V047, CVS-PL-V090, CVS-PL-V091, CVS-PL-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 that 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 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 that these valves are part of an augmented inspection program. Westinghouse stated that Note 3 in Table 3.9-16 would be revised to remove the discussion of probabilistic risk assessment for the ADS valves. Westinghouse noted that squib valves are IST Category D valves in the ASME OM Code and do not require position verification testing per ASME OM Code, Table ISTC-3500-1, "Inservice Test Requirements." Westinghouse stated that 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 that several aspects of this RAI response needed to be clarified and tracked this item as Open Item OI-SRP3.9.6-CIB1-09. Westinghouse addressed this open item in its letter dated January 26, 2010. First, Westinghouse stated that ~~To resolve this RAI, the following aspects of the Westinghouse response need to be addressed: (a) reliance on Appendix J testing for~~ 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 that 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-CIB1-10**.~~to address any additional safety functions of these valves (for example, Section 4.4.5 in NUREG-1482, Revision 1, indicates that containment isolation valves may have additional safety functions that might not be adequately addressed by Appendix J leakage testing); (b) plans for Note 3 that do not appear in the indicated Table 3.9-16 changes; and (c) intent of the response to item m, which addresses IST Category A valves although the CVS valves in the referenced Note 32 were reclassified as Category B or C valves. This concern is identified as **Open Item OI-SRP3.9.6-CIB1-09**.~~

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 that 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 that 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 that 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 that 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 that 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 ~~for~~ 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 open and confirmatory items. The NRC staff will review the operational program aspects regarding the functional design, qualification, and IST programs for safety-related 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 that 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 that 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 thẹ 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 that 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 that are required to meet Class 1E requirements are evaluated for environmental conditions including normal operation and postulated accident conditions. Accordingly, the staff concludes that 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 Certified Seismic Design Response Spectra (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 hard rock high frequency (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 staff Requirements Memorandum 93-087 issued on July 21, 1993. The regulatory basis for Section 3.10 of the AP1000 DCD is documented in NUREG-1793. It is acceptable for Westinghouse not to use the Experience-Based Qualification Method for AP1000 mechanical and electrical equipment.

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 that 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 that 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 that 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 that equipment.

In Section 6.4.5 of TR-115 for the Screening Process, Westinghouse concluded that 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 Hard Rock High Frequency (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 that above 50 Hz, the zero period acceleration (ZPA) regions of the response spectra are being approached. Westinghouse further stated that 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 that 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. In the May 28, 2008, response, Westinghouse stated that 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 ML082350116). Westinghouse stated that 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 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 that 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 that 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 that 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 that Table 3.11-1 of AP1000 DCD Revision 16 was reviewed to verify that 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 that 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 that those electronic components in question are included in those 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 that 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 that seismic qualification to the AP1000 certified design ISRS envelops the HRHF seismic inputs for most applications. Westinghouse further stated that 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 that 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 that these support loads are reduced for HRHF evaluation when compared to the CSDRS analysis. The comparison also indicates that CSDRS would control the cyclic loading demand. Westinghouse further stated that 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 that 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 that the mechanical components listed in Table 3.2-3 of the AP1000 DCD, Tier 2 that must be designed for the SSE are those classified as Seismic Category, I and II. Among those equipment and components, Westinghouse stated that 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 that 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 that 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 that 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/NRC 2503, dated May 27, 2009 (ADAMS Accession Number ML091520090). Westinghouse stated that it is performing seismic qualification of safety-related structures, systems and components (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 that 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, that 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 that Westinghouse has adequately addressed the questions raised in this RAI. The staff has also verified that 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 that 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 that the five OBE (one-half SSE) and a minimum of one SSE AP1000 ISRS test runs preceding the HRHF screening test were performed in compliance with IEEE Std 344-1987. The staff understands that the same specimen is used for all these test runs. Westinghouse also indicated that 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 that the AP1000 safety-related equipment will be ~~seismiclly~~seismically- qualified to the AP1000 Certified Seismic Design (CSD) In-Structure Response Spectra (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 that 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 that 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 Nuclear Island 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 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.

The staff has reviewed the Westinghouse's responses as stated above. The staff concludes that 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 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.

3.10.3 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 that the changes to Section 3.10 of AP1000 DCD Revision 17 are acceptable subject to **Confirmatory Item CI-SRP3.10-EMB-10**. The staff finds that the AP1000 design provides adequate assurance that AP1000 Seismic Category I equipment will function properly under the effects of earthquake motions, and that 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 that 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 that specified in NUREG-1793.

In NUREG-1793, the staff concluded that 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 that a COL applicant referencing the AP1000 design needs to fully describe EQ and other operational programs as defined in Commission Paper SECY-05-197, “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 a~~A~~pplicant may rely on information in the applicable Design Certification DCD 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 a~~A~~pplicants 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 that 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 open and confirmatory items 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 ~~and other~~ operational program. The NRC staff will evaluate the full description of the EQ Program provided by a COL applicant during review of the COL application consistent with RG 1.206 and NRC Standard Review Plan (SRP) Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment.” ~~s in support of its COL application.~~

Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” in AP1000 DCD Tier 2 presents information to demonstrate that 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 that 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 requested that 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 that 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 ~~onsite review~~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 offices in Monroeville, PA. The staff found that 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 that 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 open items (ADAMS Accession Number ML083110154). The audit response was tracked as Open Item OI-SRP3.11-CIB1-01. In Section 3.9.6 of this SER, the NRC staff discusses its review of the planned actions to address the audit findings provided in a letter from Westinghouse dated January 26, 2010. In a letter dated February 23, 2010, Westinghouse provided its response to Open Item OI-SRP3.11-CIB1-01. In particular, Westinghouse stated that the valve design specifications indicate that active valves will be qualified in accordance with ASME Standard QME-1, and that the specifications will be clarified that any existing testing must fully satisfy ASME QME-1-2007. Westinghouse also provided planned changes to the AP1000 DCD to ensure consistency in the EQ provisions and tables. **Open Item OI-SRP3.11-CIB1-01** will remain open pending a follow-up audit to review the AP1000 design specification changes, including mechanical equipment EQ provisions. **Confirmatory Item CI-SRP3.11-CIB1-01** will be used to track the planned AP1000 DCD changes related to the EQ provisions and tables.~~At the conclusion of the onsite review, the staff provided comments on the AP1000 design and procurement specifications, and Westinghouse indicated that those comments would be addressed in a future revision to the specifications. The staff also identified several items that remain open from the onsite review that are specified in Section 3.9.6 of this SER. The staff identified this concern as **Open Item OI-SRP3.11-CIB1-01.**~~

Section 3.11.5, "Combined License Information Item for Equipment Qualification File," in Revision 17 to the AP1000 DCD states that 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 that Westinghouse 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 that 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 that 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 to clarify the ~~be acceptable in describing 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 that 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 that Revision 17 to the AP1000 DCD continues to satisfy the NRC regulations applicable to the EQ Program for electrical and mechanical equipment within the scope of the EQ Program for ~~important to safety and safety-related mechanical equipment used in~~ 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 Program for electrical ~~equipment important to safety and safety-related~~and mechanical equipment to be used at an AP1000 nuclear power plant, pending resolution of OI-SRP3.11-CIB1-01. Further, the staff concludes that the AP1000 DCD changes related to the EQ 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 provide 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.” The following evaluation discussed the results of the staff’s review.

3.12.1.1 Design Specification and Reports

The staff reviewed the proposed change regarding completion of piping design reports as stated in Subsection 3.9.8.2 of AP1000 DCD Revision 17.

In Subsection 3.9.8.2 the applicant stated that "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 that 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 that "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 that 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. If this was the case, the staff requested that Westinghouse revise the DCD to reflect the design completion. Otherwise Westinghouse should provide justification for changing the COL Item from one that would be completed by the COL applicant to a COL holder item and 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 that it was the intention of Westinghouse to have the design documents for the risk-significant lines 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 that 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 that the schedule for the risk-significant piping packages would be completed by June 30, 2009, in order to resolve the piping DAC. This remains an Open Item pending NRC review of the completed piping packages. **The staff identified this concern as Open Item OI-SRP3.12-EMB-4**.

Subsection 3.9.8.2 of DCD Revision 17 proposed that the following activities were to be completed by the COL holder:

"Reconciliation of the as-built piping (verification of the thermal cycling and stratification loading considered in the stress analysis discussed in Subsection 3.9.3.1.2) is completed by COL holder after the construction of the piping systems and prior to fuel load."

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 ML081780174), the applicant responded that 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 that "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 that 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 that the COL Applicant will implement a benchmark program as described in Subsection 3.9.1.2 if a piping computer program other than 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 that 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 that 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, which stated that all piping analysis performed for the 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 proposed change to Appendix 3I, and its supporting document, TR-115, “Effects of High Frequency Seismic Content on SSCs,” APP-GW-GLR-115, October 2007. Subsection 3.7.3.7 of the AP1000 DCD stated that “For the seismic response spectra analyses, the ZPA cut-off frequency is 33 Hz.” In Appendix 3I, Figures 6.3.2.1-3 and 6.3.2.2-3 showed that the beginning of the rigid region occurs at a much higher frequency than 33 Hz for both AP1000 design and hard rock high frequency (HRHF) design. The analysis in TR-115 does not appear to use 33 Hz as the ZPA cut-off frequency as defined in the DCD, resulting in inconsistency, which needs to be addressed.

By letter dated June 6, 2008 (ADAMS Accession Number ML081620075), the applicant stated that both the AP1000 design response spectra and HRHF spectra PIPESTRESS models were built with the automatic mass modeling option set to 99 Hz. The comparison was based on the same model with 99 Hz cut-off. The staff noted that the comparison shall evaluate the effects between AP1000 CSDRS and HRHF seismic input case. By letter dated August 29, 2008

(ADAMS Accession Number ML082480501), the applicant provided a revised response, which stated that the cutoff frequency for the AP1000 CSDRS-based case was 33 Hz and the cutoff frequency for the HRHF-GMRS case was the ZPA frequency and that the issue is addressed in Revision 1 of TR-115 (ADAMS Accession Number ML082940062). On the basis that the comparison is made between the Standard AP1000 design with 33Hz cutoff frequency as defined in the DCD and HRHF seismic input with actual ZPA as cutoff frequency, the staff finds this acceptable.

In RAI-SRP3.12-EMB-2, the staff requested that the applicant provide a justification for stating that the two piping systems chosen are representative for all piping systems. For example, the floor response spectrum at Elevation 134.25' of the Containment Building has more exceedance in the high frequency region than those used in the demonstration. Floor Response Spectra should be taken into account in determining which packages envelop the complete piping design.

In letter dated June 6, 2008 (ADAMS Accession Number ML081620075), the applicant responded that looking at AP1000 vs. HRHF response spectra comparisons reveals several locations in which exceedances occur at individual node locations. However, when these comparisons are taken over multiple nodes that encompass a piping system, the AP1000 design response spectra tend to bound the HRHF response spectra. For example, the piping package APP-PXS-PLA-030 has node locations with exceedances as high as 200 percent in the high frequency region. When these nodes are compared with other locations, these exceedances are muted, if not eliminated. Therefore, exceedances of the AP1000 design response spectra by the HRHF spectra at individual node locations do not properly reflect the response spectra applied to these piping systems for qualification. The applicant stated that both exceedances in response spectra and high frequency participation are required to show significant effects. Exceedances in the high frequency region are insignificant without participation. The applicant also stated that the occurrence of the two is rare for the AP1000. The exceedance and participation comparisons of the piping system APP-PXS-PLA-030 seems like a poor candidate. However, in comparisons with other piping packages it is not a poor candidate for analysis, by contrast, a strong one. Also, the comparison was not limited to 40 packages: many more isometrics were reviewed. The staff noted that the comparison basis for the cutoff frequency and inconsistent methods as indicated in the RAI-SRP3.12-EMB-1 and EMB-3 requires additional clarification in order to demonstrate the comparison.

By letter dated August 29, 2008 (ADAMS Accession Number ML082480501), the applicant provided a revised response. In the revised response, the applicant stated that both the AP1000 Certified Seismic Design Response Spectra (CSDRS) and HRHF Ground Motion Response Spectra (GMRS) have been enveloped across entire building elevations. This is not only a conservative approach, but this also eliminates concerns of building location as the spectra are representative of an entire elevation. The staff reviewed the applicant's response and determined that the applicant is enveloping across entire building elevations to perform the analysis is conservative.

In RAI-SRP3.12-EMB-3, the staff questions why there were two inconsistent methods (enveloped vs. multiple-level response) used for piping analysis comparison. In TR-115, the applicant used enveloped floor response spectra for AP1000 design and multiple level response spectra for HRHF analysis, respectively. This comparison does not 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. Table 3.7.1-1 of AP1000 DCD states that independent support motion response spectra (i.e., multiple-level response spectra) analysis shall use either 2 percent or 3 percent damping, not the 5 percent damping used in TR-115. The staff requested that the applicant provide a comparison between the AP1000 design and the HRHF analysis using the methodology called out in the DCD.

By letter dated June 6, 2008 (ADAMS Accession Number ML081620075), the applicant indicated that the TR-115 study was to remove excessive conservatism to reflect the HRHF realistic conditions. The applicant also stated that the study does not require the design analyses and high frequency analyses to be identical to be meaningful. The staff did not concur and requested the applicant to provide HRHF analyses using the method called out in the DCD.

By letter dated August 29, 2008 (ADAMS Accession Number ML082480501), the applicant stated that both AP1000 CSDRS and HRHF GMRS analyses used uniform support motion methodology, which allows damping values of four percent and five percent to be used. Because the analyses used consistent method and damping values for comparison, the staff finds that the applicant's comparison basis is acceptable.

The staff reviewed DCD Revision 17, and Revision 1 of TR-115, which described the HRHF susceptibility study, package consideration and summarized the comparison between CSDRS and HRHF for the AP1000 design. The staff guidance and position on addressing high frequency ground motions is documented in COL/DC-ISG-1, "Interim Staff Guidance (ISG) on Seismic Issues Associated with High Frequency Ground Motion in Design Certificate and Combined License Applications" (ADAMS Accession Number ML081400293). The applicant listed the reasons for susceptibility of the analysis packages to high frequency excitation in Table 6.3.1-1 and provides further evaluation for two piping packages, APP-PXS-PLA-030 and APP-RNS-PLA-170. The staff finds that the applicant's evaluation and results meet the intent of the ISG. On this basis, the staff finds the applicant's package consideration and selected analysis candidates acceptable.

The staff reviewed the results summary for the APP-PXS-PLA-030 piping package. The results indicate that no points show an increase in pipe stress and support loads show no increase. The results demonstrate that the CSDRS analysis is more conservative than the HRHF GMRS analysis for the APP-PXS-PLA-030 piping package. The staff also reviewed the results summary for the APP-RNS-PLA-170 piping package. The results indicated that the majority of all points showed a decrease or no change in the stress ratio. For the points that did show stress ratio increases, the stress ratios were already low and remained low (<0.5). The largest stress ratio increase was from .386 to .454. The few resultant support and anchor loads increases were at points with low loads. At other points with higher loads, increases were within 10 percent. These small increases could be reduced or eliminated with more complex analysis techniques. These techniques further show that HRHF has minimal impact on piping stresses. These techniques include: a) use of multiple response spectra, b) use of more selective input response spectra instead of enveloping entire floors, which is the practice used for the AP1000 design basis analysis and c) use of time history analysis. The staff determined that those small increases in the stress and support load are not the controlling points. The controlling points are those high stress points and high support load points. The staff concluded that these techniques can further reduce the analysis result.

On the basis mentioned above, the staff concluded that the proposed HRHF response spectra, provided in the TR-115 Revision 1, has minimal impact on piping stresses and the AP1000 CSDRS design can be used for those HRHF sites which have response spectra bounded by the proposed response spectra.

3.12.1.4 Reactor Coolant Loop (RCL) 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 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 .**

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, pending resolution of the identified open items, the staff concludes that supports of piping systems important to safety are designed to quality standards commensurate with their importance to safety. 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.

References

1. Technical Report 103 (APP-GW-GLN-019, Revision 2), September 2007
2. Technical Report 106 (APP-GW-GLN-106, Revision 1), September 2007
3. 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 (ADAMS Accession Number ML083520635)
4. Memorandum from M. Patterson to Eileen McKenna dated March 17, 2009 SUBJECT: NRC On-Site Review of the Integration of RTNSS with Classification Process and Chapter 19 FSER Open Items in AP1000 (ADAMS Accession Number ML090640216)

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