14. VERIFICATION PROGRAMS

14.2 Initial Plant Test Program

14.2.9 Preoperational Test Abstracts

14.2.9.1 Introduction

Westinghouse (the applicant) has proposed significant changes to the following items in the AP1000 design control document (DCD) related to preoperational testing: (1) squib valves associated with passive core cooling (DCD Section 14.2.9.1.3); (2) control rod drive system (CRDS) (DCD Section 14.2.9.8); and (3) main alternating current (ac) power testing (DCD Section 14.2.9.2.15).

14.2.9.2 Evaluation

The Nuclear Regulatory Commission (NRC) staff sent the applicant a request for additional information (RAI) regarding the passive core cooling system test description in Section 14.2.9.1.3 of the AP1000 DCD. The NRC staff noted in RAI-SRP14.2-CQVP-10, that under the "General Test Methods and Acceptance Criteria," Item (t), the applicant described the testing of the squib valves as they relate to verification of the passive core cooling system safety injection function. The applicant stated that this test does not have to be performed in the plant. The applicant added this last sentence to the test abstract as part of Revision 16 of the AP1000 DCD. In the RAI, the staff requested that the applicant provide justification for this change. In its July 11, 2008, response to this RAI, the applicant stated that the last sentence of this section was added as an editorial change to clarify that this testing could be done without causing the risk of an actual safety injection into the core. Additionally, the applicant stated that the reliability of these valves could be verified without the valves actually being tested in the operating passive core cooling system.

The staff reviewed the applicant's response to this RAI and determined that this change does not affect the test of the squib valves consistent with regulatory guide (RG) 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 3, and the performance of these valves could be adequately verified through qualification testing. On this basis, the staff concludes that the passive core cooling system test adequately addresses proper operation of the squib valves and verifies adequate safety injection, and is, therefore, acceptable. This resolves RAI-SRP14.2-CQVP-10.

In RAI-SRP14.2-CQVP-9, the staff requested additional information regarding the CRDS test description in Section 14.2.9.1.8 of the AP1000 DCD, Revision 16. The staff noted that Section 14.2.9.1.8 of the AP1000 DCD, Revision 16, CRDS, stated that, as a prerequisite for the control rod drive mechanism cooling test, "the plant is at or near normal operating temperature and pressure, and post-core hot functional testing is in progress." The staff noted that the applicant added the word "post-core" to the test abstract as part of Revision 16 of the AP1000 DCD and asked that the applicant justify this change. In its July 11, 2008, response to this RAI, the applicant stated that the addition of the word "post-core" to modify the hot functional testing of the control rod drive mechanism was an editorial change to clarify the fact that this test can only occur after the core is loaded. The staff reviewed the applicant's response to this RAI and determined that this change clarifies the prerequisites of this test, does not affect the test abstract for the CRDS, and is consistent with the test recommended in

RG 1.68. Therefore, the staff finds this change to be acceptable. This resolves RAI-SRP14.2-CQVP-9.

In test abstract 14.2.9.2.15, "Main AC Power System Testing," the staff noted that the applicant included additional verification activities under "General Methods and Acceptance Criteria" for the bus transfer schemes as part of the test activities associated with the main ac power system. The applicant modified this test abstract to ensure that appropriate testing of the bus transfer schemes occurs during the preoperational phase of the initial test program. Section 8.3.1 of the AP1000 DCD provides details regarding the ac power system and the function of the bus transfer schemes.

The staff reviewed the applicant's proposed change to test abstract 14.2.9.2.15 and determined that this change provides a means to verify proper operation of the automatic and maintenance bus transfer schemes, does not affect the test of the main ac power system, and is consistent with the guidance in RG 1.68. On this basis, the staff concluded that the change is acceptable.

The staff notes that the applicant introduced several other changes to the preoperational test abstracts in Revision 17 of the AP1000 DCD. Upon review, the staff found that these changes were consistent with plant design changes and equipment naming conventions and have no significant impact on preoperational testing. Therefore, these changes are acceptable.

14.2.9.3 Conclusion

The staff reviewed the proposed changes to the preoperational test program described in Section 14.2.9 of the AP1000 DCD. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Section XI, "Test Control," requires that the applicant establish a test program which will ensure that all of the testing required to demonstrate that structures, systems, and components (SSCs) will perform satisfactorily in service, and will be performed in accordance with written test procedures which incorporate the requirements and acceptance limits in applicable design documents. On the basis of this review, the staff concludes that the proposed changes satisfy the requirements of Appendix B to 10 CFR Part 50, Section XI.

14.3 Tier 1 Information

14.3.2 Inspection, Test, Analysis, and Acceptance Criteria (ITAAC)

A number of changes to ITAAC were proposed in revisions to the AP1000 DCD. The staff's evaluations of the proposed changes to ITAAC (which are Tier 1 information) appear in those safety evaluation report (SER) sections where the subject SSCs are discussed. For example, with regard to changes in AP1000 DCD Tier 1, Table 2.1.1-1 concerning the fuel handling machine, these changes are evaluated in Section 9.1 of this report.

14.3.3 Design Acceptance Criteria (DAC)

Changes to design acceptance criteria (DAC) were proposed in Revisions 16 and 17 of the AP1000 DCD. The staff's evaluations of the proposed changes to DAC (which are Tier 1 information) appear in those SER sections where the subject SSCs are discussed. For example, the Tier 1 information found in AP1000 DCD Tier 1, Section 2.5.2, "Protection and Safety Monitoring Systems," Item 11, addresses the hardware and software development

process for the design, testing and installation of instrumentation and control (I&C) equipment. Changes to this Tier 1 information are evaluated in Section 7.2 of this report.

14.3.5 Changes to Tier 1 Information

This section addresses the proposed changes to the AP1000 DCD Tier 1 information in the application for amendment to the design certification (DC), as supplemented. Tier 1 information includes the following:

- definitions and general provisions
- design descriptions
- ITAAC
- significant site parameters
- significant interface requirements

The Tier 1 information is derived from the AP1000 DCD Tier 2 information. This SER evaluates the proposed changes to the Tier 1 information in those SER sections in which the associated changes to the Tier 2 information are located, except as described below.

14.3.5.1 Evaluation

Section 1.1 of the AP1000 DCD Tier 1 provides the current definition of "as-built" as follows:

> As-built means the physical properties of a structure, system, or component (SSC) following the completion of its installation or construction activities at its final location at the plant site.

This definition will be used in implementing the ITAAC verification process. This definition intends that the determination of whether an SSC meets the acceptance criteria in the respective ITAAC be performed at its final, in-place location. This approach meets the intent of the ITAAC requirement in 10 CFR 52.97(b), "Issuance of combined licenses."

In a letter dated June 13, 2008, the applicant submitted AP1000 DCD Impact Document APP-GW-GLE-007, Revision 0, "ITAAC Changes." One of the proposed changes to the ITAAC would change the definition of "as-built." The proposed change would add the following sentence to the definition of "as-built":

> Determination of physical properties of the as-built structure, system, or component may be based on measurements, inspections, or tests that occur prior to installation provided that subsequent fabrication, handling, installation, and testing does not alter the properties.

This proposed change to the definition of "as-built" concerned the staff because the AP1000 ITAAC were developed with the expectation that verifications would be performed in the final, in-place location of the SSC. Also, the shipping, storage, handling, and installation performed after vendor fabrication and prior to the final, in-place location can damage an SSC. The staff raised these concerns during discussions with representatives of the Nuclear Energy Institute (NEI).

On August 1, 2008, NEI submitted NEI 08-01, Revision 0, "Industry Guidelines for ITAAC Closure Process under 10 CFR Part 52," a draft of which was the basis for the proposed change to the definition of "as-built" in APP-GW-GLE-007, Revision 0. Section 3.1.4 of NEI 08-01 includes the following statement to clarify the definition of "as-built":

> Many ITAAC require verification of "as-built" SSCs. However, some of these ITAAC will involve measurements and/or testing that can only be conducted at the vendor site due to the configuration of equipment or modules or the nature of the test (e.g., measurements of reactor vessel internals). For these specific items where access to the component for inspection or test is impractical after installation in the plant, the ITAAC closure documentation (e.g., test or inspection record) will be generated at the vendor site and provided to the licensee.

The staff understands that it may be impossible to perform some ITAAC verifications of an SSC in its final, in-place location. Therefore, the staff agrees with the NEI proposal to modify the definition of "as-built" with the clarification provided by the above statement and additional documentation demonstrating that subsequent fabrication, handling, installation, and testing did not alter the properties of the SSC.

In RAI-SRP14.3-NWE2-01, the staff requested that the applicant incorporate the above clarification into the AP1000 definition of "as-built" since it significantly restricts the completion of ITAAC at a vendor's site. In its response to RAI-SRP14.3-NWE2-01, dated September 9, 2008, the applicant declined to incorporate the subject clarification into the definition of "as-built." In a December 9, 2008, revised response to RAI-SRP14.3-NWE2-01, the applicant proposed to incorporate the subject clarification in Tier 2. The staff's position is that this approach would provide too much latitude in application of the subject clarification and that the clarification should be incorporated into the definition itself in Tier 1 of the AP1000 DCD. This was identified as Open Item OI-SRP14.3-NWE2-01.

The industry and the NRC subsequently agreed to augment the NEI 08-01 definition of "as-built" to match the definition agreed to between the NRC and the industry at the December 17, 2009, Category 3 public meeting on ITAAC maintenance. The following definition is considered acceptable:

> **As-built** means the physical properties of a structure, system, or component following completion of its installation or construction activities at its final location at the plant site. In cases where it is technically justifiable, determination of physical properties of the as-built structure, system, or component may be based on measurements, inspections, or tests that occur prior to installation, provided that subsequent fabrication, handling, installation, and testing do not alter the properties.

In its February 19, 2010, response to RAI-SRP14.3-NWE2-01, the applicant adopted the above definition for use in Section 1.1 of Tier 1 of the AP1000 DCD. In addition, in the February 19, 2010, response the applicant indicated that, as the result of an extensive review of the ITAAC in Revision 17 of the AP1000 DCD, to ensure that those ITAAC utilizing the definition of the term "as-built" will be implemented consistent with the new definition, a number of changes to the ITAAC would be necessary. The staff has reviewed the proposed changes to the ITAAC that use the definition of "as-built" and concludes that these changes do not adversely affect inspectability or any technical aspect of the ITAAC and that these changes are acceptable.

The staff confirmed that Revision 19 to the AP1000 DCD contains the acceptable definition of "as-built" and associated ITAAC changes.

14.3.5.2 Conclusion

The proposed change to the DCD, to include the definition of "as-built" and associated changes to the ITAAC, are acceptable.

14.3.6 Design Acceptance Criteria/ITAAC Closure Process

In an April 1, 2010, Revision 1 response to Open Item OI-SRP3.12-EMB-4, the applicant proposed to include Appendix 14.3A in Tier 2, Chapter 14 of the AP1000 DCD as generic guidance for the resolution of DAC in the DCD. In a proposed Tier 2 Appendix 14.3A, the applicant discussed the options for resolving DAC following certification of the design: (1) through amendment of the DC rule; (2) through the combined operating license (COL) application review process; or (3) through ITAAC after COL issuance. For example, for the piping DAC, the applicant proposed that a COL information item be included in the DCD to resolve the piping design outside of the DC amendment. Closure of piping DAC is further discussed in Section 3.12 of this report.

The standard approach outlined above is voluntary on the part of each licensee referencing the standard AP1000 design. The process envisions an NRC review, inspection, or audit of the DAC completion that applies the "one issue, one review, one position" concept as discussed in RG 1.206, "Combined License Applications for Nuclear Power Plants." Section C.III.5, to DAC resolution for the reference (first) AP1000 plant and to subsequent AP1000 plants. A COL applicant can apply this standard approach to each of the AP1000 design areas that include DAC (i.e., piping design, digital I&C design, and HFE design). When DAC applies, the process indicates the COL applicant is to provide ITAAC and closure schedule indicating the approach to be followed for the site. The staff finds that this standard approach is consistent with the NRC policy for a design-centered-review approach and the regulations and is, therefore, acceptable.

The staff confirmed that Revision 19 to the AP1000 DCD includes the acceptable standard approach.

14.4 Combined License Applicant Responsibilities

14.4.1 Test Specifications and Procedures

14.4.1.1 Introduction

In a letter dated September 22, 2006, and as supplemented in a letter dated May 11, 2007, the applicant submitted technical report (TR)-71A, APP-GW-GLR-037, "AP1000 Test Specifications and Procedures," for NRC review and approval. The applicant requested resolution of COL Information Item 14.4-2 based on the information provided in TR-71A. Section 14.4 of the AP1000 DCD lists this COL information item and assigns the COL applicant the responsibility of providing test specifications and test procedures for preoperational and startup tests to the NRC for review and approval.

14.4.1.2 Evaluation

In Section 14.4.3 of the AP1000 DCD, Revision 15, the applicant assigned the responsibility for the development of preoperational and startup test specifications and procedures to the COL applicant. Specifically, COL Information Item 14.4-2 states the following:

> The COL applicant is responsible for providing test specifications and test procedures for the preoperational and startup tests, as identified in Section 14.2.3, for review by the NRC.

As part of Revision 16 to the AP1000 DCD, the applicant submitted TR-71A to address COL Information Item 14.4-2. TR-71A outlines the process to be used by the applicant to develop test specifications and draft procedures and provides a list of test specifications and test procedures to be provided in draft form by the applicant to the prospective COL holder.

TR-71A documents the development process for the preparation of 88 preoperational system test specifications and 59 startup test specifications, to be followed by 289 preoperational test procedures and 59 startup test procedures. However, TR-71A does not include the actual test specifications and test procedures for NRC review and approval.

The staff determined that COL Information Item 14.4-2 calls for the actual submittal of test specifications and test procedures by a COL holder to the NRC onsite inspectors for review and approval before as-built systems and plant features are tested in the field. Furthermore, the NRC inspection staff will need to review the actual test specifications and test procedures for components and systems to be tested to verify their acceptability before COL Information Item 14.4-2 can be categorized as resolved. Accordingly, resolution of COL Information Item 14.4-2 will be subject to the NRC's construction inspection program to allow for the necessary plant as-built inspections and walkdowns. On this basis, the staff concludes that COL Information Item 14.4-2 will remain open pending submittal of the required information by the COL holder.

The staff notes that Section 14.4.2 of Revision 17 of the AP1000 DCD includes the following statement:

> The Combined License information requested in this subsection has been partially addressed in APP-GW-GLR-037 (Reference 1), and the applicable changes incorporated into the DCD. Test specifications have been developed as indicated in Reference 1 and are available for NRC onsite review at Westinghouse's offices.

The above noted statement is inconsistent with the staff's conclusion; it should be deleted from the AP1000 DCD, and COL Information Item 14.4-2 should be modified to refer the responsibility for the information item to the COL holder. This was designated as Open Item OI-SRP14.2-CQVP-12.

In its September 30, 2009 response to Open Item OI-SRP14.2-CQVP-12, the applicant stated that it would remove reference to TR-71A in a revised version of the AP1000 DCD. The applicant also proposed to remove any references to TR-71A and to restore the text that would make the COL applicant responsible for providing the necessary information. The applicant included the proposed language to modify COL Information Item 14.4-2 in Enclosure 1 of the September 30, 2009 letter. The response to Open Item OI-SRP14.2-CQVP-12 is acceptable.

The staff confirmed that a subsequent revision to the AP1000 DCD includes the correct COL information item.

14.4.1.3 Conclusion

The staff reviewed the information submitted by the applicant in TR-71A to resolve COL Information Item 14.4-2 in Section 14.4 of the AP1000 DCD, Revision 16. On the basis of this review, the staff concludes that COL Information Item 14.4-2 cannot be resolved until after the issuance of the COL. Therefore, COL Information Item 14.4-2 will reflect that required information is to be provided by the COL holder.

14.4.2 Conduct of Test Program

14.4.2.1 Introduction

In a letter dated September 26, 2006, and as supplemented by letters dated May 24, 2007, and June 19, 2008, the applicant submitted TR-71B, APP-GW-GLR-038, "AP1000 Conduct of Test Program," for NRC review and approval. The applicant requested that COL Information Item 14.4-3 be resolved based on the information provided in TR-71B. Section 14.4.3 of the AP1000 DCD lists this COL information item as follows:

> The Combined License applicant is responsible for a startup administration manual (procedure), which contains the administrative procedures and requirements that govern the activities associated with the plant initial test program, as identified in subsection 14.2.3.

14.4.2.2 Evaluation

In Section 14.2.3.1 of the AP1000 DCD, Revision 15, the applicant provided a set of administrative requirements for the conduct of the initial test program. In addition, Section 14.4 of the AP1000 DCD, Revision 15, summarized the COL applicant responsibilities associated with the development of a startup administrative manual (SAM). COL Information Item 14.4-3 required applicants referencing the AP1000 DCD to provide administrative controls for the conduct of the initial test program in the form of a SAM.

As part of Revision 16 of the AP1000 DCD, the applicant submitted TR-71B to resolve COL Information Item 14.4-3. In reviewing TR-71B, the staff noted that the applicant provided a summary overview of the administrative process and program controls to be utilized in the conduct of the AP1000 startup test program at a licensed AP1000 operational plant site.

The staff also noted that TR-71B outlined basic functional relationships, responsibilities, activities, authority, and principles of conduct for the Joint Test Working Group and other organizational groups. Additionally, TR-71B presented a general and informative description of responsibilities and activities related to the testing of power plant equipment in the period between system turnover and plant acceptance.

On the basis of its review of TR-71B, the staff identified several areas that required additional information as presented in RAI-SRP14.2-CQVP-01 through RAI-SRP14.2-CQVP-08 and RAI-SRP14.2-CQVP-11. To this end, the staff requested that the applicant enhance the proposed program description and amplify the administrative requirements in TR-71B consistent with the guidance in RG 1.68 and Section 14.2, "Initial Plant Test Program—Design Certification

and New License Applicants," of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." Specifically, the staff requested that the applicant provide not only a list of activities that will be controlled during the conduct of the initial test program, but also a description of how these activities will be implemented and controlled. The staff also requested that the applicant justify the use of references to other documents in TR-71B that were not currently under review by the staff. In addition, the staff provided the applicant a set of administrative controls that needed to be described as part of the development of the AP1000 SAM.

In its June 19, 2008, response, the applicant provided Revision 2 of TR-71B. The staff reviewed Revision 2 of TR-71B and noted that the applicant had revised the organizational structure in charge of the initial test program and enhanced the description of the administrative controls for the startup testing phase of the initial test program. The applicant also provided partial responses to some of the RAIs and, in certain areas, did not provide a response at all. The applicant stated in its response letter that Revision 2 of TR-71B incorporated the responses to the RAIs and that the initial test program was developed in conformance with RG 1.68, Revision 2, issued August 1978, and Section 14.2 of NUREG-0800, Revision 2, issued July 1981, which is the certified design regulatory basis.

Because COL applicants are incorporating TR-71B by reference in their applications, the staff determined that the applicant's responses created a conflict between the information provided by the applicant and that required to be submitted by COL applicants. In addition, the staff determined that COL Information Item 14.4-3 calls for the actual submittal of a SAM describing the methods and practices that would govern the initial test program at AP1000 sites. This SAM should provide controls for the conduct of the initial test program consistent with the general criteria in RG 1.68, Revision 3, and Section 14.2 of NUREG-0800, Revision 3. On this basis, the staff determined that the existing content and structure of TR-71B does not meet the guidance applicable to COL applicants.

Since the applicant has not provided the necessary information consistent with COL Information Item 14.4-3, the staff concludes that COL Information Item 14.4-3 will remain open in the AP1000 DCD pending submittal of the required information by COL applicants referencing the AP1000 design.

The NRC staff notes that Section 14.4.3 of Revision 17 of the AP1000 DCD includes the following statement:

> The Combined License information requested in this subsection is partially addressed in APP-GW-GLR-038, (Reference 2), and the applicable changes are incorporated into the DCD.
>
> The program management description for the process to develop the AP1000 Startup Administrative Manual is delineated in APP-GW-GLR-038, (Reference 2).

The above-noted statement is inconsistent with the NRC staff's conclusions; it should be deleted from the DCD, and COL Information Item 14.4-3 should be restored to the original COL information item commitment noted in Section 14.4.3 of Revision 15 of the AP1000 DCD. This was identified as Open Item OI-SRP14.2-CQVP-13.

In its September 30, 2009 response to Open Item OI-SRP14.2-CQVP-13, the applicant stated that it would remove reference to TR-71B in a revised version of the AP1000 DCD. The

applicant also proposed to remove any references to TR-71B and to restore the text that would make the COL holder responsible for providing the necessary information. The applicant included the proposed language to modify COL Information Item 14.4-3 in Enclosure 1 of the September 30, 2009 letter. The response to Open Item OI-SRP14.2-CQVP-13 is acceptable. The staff confirmed that a subsequent revision to the AP1000 DCD contains the correct COL information item.

14.4.2.3 Conclusion

On the basis of its review of TR-71B, the staff concludes that it lacks the elements that are necessary for a SAM. Furthermore, it is inconsistent with the guidance provided in RG 1.68, Revision 3, and Section 14.2, Revision 3, of NUREG-0800. Therefore, COL Information Item 14.4-3 will remain a responsibility of the COL applicant.

14.4.3 First-Plant-Only and Three-Plant-Only Tests

14.4.3.1 Introduction

In a letter dated June 5, 2006, the applicant submitted TR-6, "AP1000 As-Built COL Information Items," for NRC review and approval. TR-6 identified COL information items that required as-built information or conditions to be completed.

In its request, the applicant proposed a change to COL Information Item 14.4-6, "First-Plant-Only and Three-Plant-Only Tests," in order to clarify that the test requirements apply to a COL holder rather than a COL applicant.

14.4.3.2 Evaluation

COL Information Item 14.4-6 is associated with tests that must be completed only on the first plant or the first three plants. For COL Information Item 14.4-6, the following revision was underlined in TR-6:

> [The COL applicant or holder for the first plant and the first three plants will perform the tests listed in subsection 14.2.5. For subsequent plants, the COL applicant or licensee shall either perform the tests listed in subsection 14.2.5 or shall provide a justification that the results of the first-plant-only tests or first-three-plant tests are applicable to subsequent plants.]*
>
> The Combined License holder will perform the tests or provide the information defined above prior to fuel load.

The staff concluded that a COL holder can only perform these tests after the plant is essentially complete. The staff found this change to COL Information Item 14.4-6 acceptable for five preoperational tests of the nine tests described in AP1000 DCD Section 14.2.5, "Utilization of Reactor Operating and Testing Experience in the Development of Test Program." However, the staff also found that the following tests described in DCD Section 14.2.5 are performed after initial fuel load and during the low-power and power ascension test phase:

- Section 14.2.10.3.6, "Natural Circulation"
- Section 14.2.10.3.7, "Passive Residual Heat Removal Heat Exchanger"

- Section 14.2.10.4.6, "Rod Cluster Control Assembly Out-of-Bank Measurements"
- Section 14.2.10.4.22, "Load Follow Demonstration"

On the basis of this finding, the staff requested that the applicant revise COL Information Item 14.4-6 to state that the COL holder will perform these tests after initial fuel load and during the low-power and power ascension test phase of the initial test program.

In its February 1, 2007, letter, the applicant revised TR-6 and COL Information Item 14.4-6 to read as follows:

> The Combined License holder(s) for the first AP1000 plant (or first three plants) available for testing will perform the tests defined during the preoperational and startup testing as identified in Subsections 14.2.9 and 14.2.10. Combined License holders referencing the results of the tests will provide the report as necessary. The schedule for providing this information will be provided prior to preoperational testing.

The change proposed by the applicant clarifies the COL holder's responsibility, in contrast to the previous assignment of responsibility to either the COL applicant or holder, for performing first-plant-only or three-plant-only tests or providing suitable justification for not performing these tests before the start of preoperational testing.

On this basis, the staff determined that this revision to COL Information Item 14.4-6 is acceptable.

14.4.3.3 Conclusion

The staff reviewed the information submitted by the applicant in TR-6 and concluded that the changes adequately clarified the timing of the testing of first-plant-only and three-plant-only tests. On the basis of this review, the staff concludes that the changes proposed by the applicant are acceptable.

15. TRANSIENT AND ACCIDENT ANALYSES

15.1 Introduction

In Chapter 15, "Accident Analyses," of the design control document (DCD), Westinghouse, (the applicant), described the safety analyses of various design-basis transients and accidents. The results of these analyses demonstrate conformance of the AP1000 design with the acceptance criteria. The acceptance criteria for the design-basis events are based on meeting the relevant requirements and general design criteria (GDC) specified in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities." The specific acceptance criteria for each event appear in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," issued March 2007. The acceptance criteria include GDC 10, "Reactor Design," pertaining to the specified acceptable fuel design limits, and GDC 15, "Reactor Coolant System Design," pertaining to the design conditions of the reactor coolant pressure boundary (RCPB). These criteria ensure that these limits are not exceeded during any conditions of normal operation, including anticipated operational occurrences (AOOs). For postulated loss-of-coolant accidents (LOCAs), the applicant analyzed various break sizes and locations to show compliance with the emergency core cooling system (ECCS) performance acceptance criteria specified in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

Various sections and tables in Section 15.0 of the DCD describe the transient and accident events analyzed in Chapter 15 and the computer codes used and assumptions made in the safety analyses of these events. The assumptions relate to the design plant conditions, initial conditions, protection and safety monitoring system (PMS) setpoints and time delays to the reactor trip functions, the actuation of engineered safety features (ESFs), and the plant systems and components available for the mitigation of transients and accidents.

The proposed changes to the DCD include numerous changes to Section 15.0. The sections below describe the specific changes and their evaluation by the staff of the U.S. Nuclear Regulatory Commission (NRC).

15.1.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.1.0.3.1 Power Measurement Uncertainty

The initial conditions assumed in the existing Chapter 15 safety analyses include a ±2-percent allowance for the power calorimetric error. In DCD, Section 15.0.3.2, "Initial Conditions," Table 15.0-2, "Summary of Initial Conditions and Computer Codes Used (Sheets 1–5)" (Footnote a), and Table 15.0-5, "Determination of Maximum Power Range Neutron Flux Channel Trip Setpoint, Based on Nominal Setpoint and Inherent Typical Instrumentation Uncertainties," now include the following sentence:

> The main feedwater flow measurement supports a 1 percent power uncertainty; use of a 2 percent power uncertainty is conservative.

The applicant also changed the initial thermal power assumed for the large-break LOCA analysis in Table 15.0-2 from 3,468.0 megawatts thermal (MWt) to 3,434.0 MWt. Based on the

AP1000 rated thermal power of 3,400 MWt, the initial thermal power of 3,434 MWt represents a 1-percent allowance for the power calorimetric error.

15.1.0.3.1.1 Evaluation

The review guidance in NUREG-0800 Section 15.0 states that the reviewer ensures that the application specifies the permitted fluctuations and uncertainties with reactor system parameters and assumes the appropriate conditions, within the operating band, as the initial condition for transient analysis. For analyses of postulated LOCAs, Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 specifies that an assumed power level lower than 1.02 times the licensed power level may be used, provided the proposed alternative value has been demonstrated to account for uncertainties resulting from errors in power-level instrumentation.

The existing Chapter 15 safety analyses for all transients and accidents assume the rated thermal power with a calorimetric error of 2.0 percent as the initial condition. The DCD continues to assume the 2 percent power uncertainty for all events except for the large-break LOCA; for this event, it assumes the rated thermal power with a 1 percent power uncertainty as the initial condition. The AP1000 DCD does not describe the instrumentation or methodology used for the main feedwater flow measurement, nor does it provide a basis to support the claimed 1 percent power uncertainty.

In its response to request for additional information (RAI)-SRP-15.0-SRSB-02, the applicant stated that the AP1000 will use the proven application of high-accuracy ultrasonic feedwater flow measurement and high-accuracy feedwater temperature measurement to affect a high-accuracy plant calorimetric. AP1000 licensees will calculate the plant calorimetric uncertainty and verify that the actual plant instrumentation performance is bounded by the design value of 1 percent calorimetric uncertainty. For traceability, Section 15.0.15, "Combined License Information," will include the following combined license (COL) information item:

> 15.0.15.1 Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters prior to fuel load, the Combined License holder will calculate the primary power calorimetric uncertainty. The calculations will be completed using an NRC acceptable method and confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated values.

The applicant will also revise DCD Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items," to include COL Item 15.0-1, "Documentation of Plant Calorimetric Uncertainty Methodology," as an action item for the COL holder.

The staff notes that the NRC has approved high-accuracy ultrasonic feedwater measurement instrumentation that can achieve a 1 percent power measurement uncertainty, and this instrumentation has been used as a basis for power uprates for operating plants. The staff concludes that COL Information Item 15.0.15.1 and COL Item 15.0-1 provide acceptable vehicles for the COL license holder to confirm its selected instrumentation for the main feedwater measurement, with the power calorimetric measurement uncertainty bounded by the 1 percent power uncertainty assumed in the large-break LOCA analysis.

In a subsequent revision to the AP1000 DCD, the applicant incorporated COL Information Item 15.0.15.1 in DCD Section 15.0.15 and COL Item 15.0-1 in Table 1.8-2.

15.1.0.3.1.2 Conclusion

Based on the above evaluation, the staff finds that COL Item 15.0-1 and COL Information Item 15.0.15.1 provide acceptable vehicles to confirm the 1-percent power uncertainty assumed in the large-break LOCA analysis. The uncertainty pertaining to the initial condition of the power level is, therefore, properly addressed, consistent with NUREG-0800 Section 15.0 and Appendix K to 10 CFR 50.46 regarding LOCAs. Therefore, the staff concludes that the above proposed changes are acceptable

15.1.0.3.2 Axial Power Shape

Section 15.0.3.3, "Power Distribution," in Revision 15 of the DCD states that the axial power shape used in the departure from nucleate boiling (DNB) calculation is the 1.55 chopped cosine, as discussed in Section 4.4 for transients analyzed at full power. In Revision 17 of the DCD, the applicant changed "the 1.55 chopped cosine" to "a chopped cosine."

15.1.0.3.2.1 Evaluation

Section 4.4.4.3.2, "Axial Heat Flux Distributions," of the DCD states that the reference axial shape used in establishing core DNB limits (that is, overtemperature ΔT protection system setpoints) is a chopped cosine with a peak-to-average value of 1.61. The staff finds that the applicant made the proposed change in Section 15.0.3.3 of the DCD to eliminate the specific mention of the 1.55 chopped cosine in order to correct the inconsistency with the value given in Section 4.4.4.3.2 of the DCD. The staff concludes that the proposed change is acceptable because the Chapter 15 safety analyses for the transients were performed with the axial power shape described in DCD Section 4.4.4.3.2.

15.1.0.3.2.2 Conclusion

The staff has reviewed the proposed change in Section 15.0.3.3 of the DCD to eliminate the specific value of 1.55 for the chopped cosine axial power shape. Based on the above evaluation, the staff concludes that this proposed change is acceptable because the applicant made it in order to correct an inconsistency with the value specified in DCD Section 4.4.4.3.2, which was referenced in DCD Section 15.0.3.3.

15.1.0.6 Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses

15.1.0.6.1 Change of "High-1" to "High-2" Containment Pressure for "S" Signal or Engineered Safety Feature Actuation

Revision 17 of the DCD changes Table 15.0-4a, "Protection and Safety Monitoring System Setpoints and Time Delay," from having the "S" signal trip function on high-1 containment pressure to its functioning on high-2 containment pressure. In addition, in Table 15.0-6, "Plant Systems and Equipment Available for Transient and Accident Conditions (Sheet 1 of 5)," the applicant changed the ESF actuation function credited for the steam system piping failure, feedwater system pipe break, and LOCA events from high-1 containment pressure to high-2 containment pressure.

15.1.0.6.1.1 Evaluation

The applicant made the change from "high-1" to "high-2" containment pressure in DCD Tables 15.0-4a and 15.0-6 for the "S" signal trip function and the ESF actuation function credited in the safety analyses, respectively, in order to be consistent with Technical Specification (TS) Table 3.3.2-1, "Engineered Safeguards Actuation System Instrumentation," which indicates that the safeguard actuation function 1.b is "containment pressure—high 2." Although the applicant changed the term "high-1 containment pressure" to "high-2 containment pressure," the value of the setpoint assumed in the safety analyses is not changed, and therefore the safety analyses are not affected. Since this is merely a change in terminology for consistency, the staff concludes that this change is acceptable.

15.1.0.6.1.2 Conclusion

Under 10 CFR 50.36, "Technical specifications," the plant TS must specify the limiting safety system settings of automatic protective devices. TS Table 3.3.2-1 for the AP1000 specifies the "Containment Pressure—High 2" trip setpoint for the safeguard actuation function. The staff finds the proposed change from "high-1" to "high-2" containment pressure in DCD Tables 15.0-4a and 15.0-6 to be acceptable because the applicant has made it in order to be consistent with the terminology used in the AP1000 TS, without changing the setpoint value or safety analyses.

15.1.0.6.2 Changes to Pressurizer Water Level Setpoints

In DCD Table 15.0-4a, the applicant changed the limiting setpoint for actuation of passive residual heat removal (PRHR) on the high-3 pressurizer water level from 80 percent of span to 76 percent of span. It also changed the chemical and volume control system (CVS) isolation on the high-2 pressurizer water level from 67 percent of span to 63 percent of span, and the CVS isolation on the high-1 pressurizer water level coincident with "S" signal from 30 percent of span to 28 percent of span.

15.1.0.6.2.1 Evaluation

The applicant changed the high-3, high-2, and high-1 pressurizer water level setpoints because of a design change in pressurizer dimensions. As described in technical report (TR)-36, APP-GW-GLR-016, Revision 0, "AP1000 Standard Combined License Technical Report, AP1000 Pressurizer Design," the applicant has changed the AP1000 pressurizer dimensions to accommodate the space constraint. This dimensional change reduces the height and increases the diameter of the pressurizer but maintains the same overall pressurizer volume. Therefore, in DCD Table 15.0-4a, the setpoints assumed in the safety analyses for the high-3, high-2, and high-1 pressurizer water levels change from 80 percent, 67 percent, and 30 percent to 76 percent, 63 percent, and 28 percent, respectively, to maintain the same water volumes of the corresponding setpoints. The applicant also revised the high pressurizer water level setpoints in TS Tables 3.3.1-1 and 3.3.2-1, respectively, to be consistent with the revised assumed measurement instrumentation uncertainties accounted for in the safety analysis.

In its response to RAI-TR36-012, the applicant reanalyzed Chapter 15 limiting events using the changed dimension and the revised setpoints. The results show no or minimal effects on the Chapter 15 events, and the applicable acceptance criteria for each event continue to be met. Therefore, the staff concludes that these changes are acceptable.

15.1.0.6.2.2 Conclusions

Based on the above evaluation, the staff finds that the applicant has made the proposed changes to the high-3, high-2, and high-1 pressurizer water level setpoints in DCD Table 15.0-4a and in TS Tables 3.3.1-1 and 3.3.2-1 to accommodate the pressurizer dimension change without changing the corresponding water volume. The staff concludes that these changes are acceptable because the applicant's reanalyses of the Chapter 15 limiting events using the revised dimension and setpoints show no more than minimal effects, and the applicable acceptance criteria for each event analyzed continue to be met.

15.1.0.6.3 Change to the Limiting Setpoint of the Boron Dilution Block on the Source Range Flux Doubling Function

In DCD Table 15.0-4a, the applicant changed the limiting setpoint of the boron dilution block on the source range flux multiplication (doubling) function from "1.6 over 50 minutes" to "3.0 over 50 minutes."

15.1.0.6.3.1 Evaluation

The safety analyses of the boron dilution events during shutdown operation credit the boron dilution block on the source range flux multiplication function (DCD Section 15.4.6, "Chemical Volume and Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant"). In DCD Table 15.0-4a, the applicant changed the terminology for the boron dilution block from "source range flux multiplication" to "source range flux doubling." In Section II.9, "Flux Doubling/Boron Dilution Modifications," of TR-80, APP-GW-GLR-080, "Mark-up of AP1000 Design Control Document Chapter 7," Revision 0, the applicant discussed the change for the setpoint of the source range flux doubling signal for the boron dilution block. The source range flux doubling signal is used for protection against boron dilution events during shutdown operation. If the source range neutron flux were to increase and exceed the setpoint because of a decrease in boron concentration in the reactor coolant, the boron dilution protection functions would be actuated. The applicant stated that an analysis of the source range flux doubling setpoints indicates that there is a significant likelihood that the setpoints could lead to inadvertent actuation of the boron dilution protection actions when the plant is shut down and the flux doubling function is active, even in the absence of actual changes in core neutron multiplication. This occurs because of the variability inherent in counting a discrete random process such as the leakage of neutrons from the reactor core, especially at relatively low count rates. Therefore, Revision 16 of the DCD changed the nominal setpoint from 1.6 over 50 minutes to 2.2 over 50 minutes to reduce the likelihood of inadvertent actuation of the boron dilution protection functions during normal operation.

DCD Section 15.4.6 provides the safety analyses of boron dilution events resulting from CVS malfunction during various modes of operation. The analyses for dilution during cold shutdown (Mode 5), safe shutdown (Mode 4), hot standby (Mode 3), and startup (Mode 2) take credit for the source range flux doubling signal with the increased setpoint that isolates the makeup flow to the reactor coolant system (RCS) from the demineralized water storage tank. The results indicate that automatic protective actions initiate to minimize the approach to criticality and maintain the plant in a subcritical condition.

In Revision 17 of the DCD, TS Table 3.3.2-1, the trip setpoint for function 15.a also specifies the source range flux doubling setpoint of 2.2 over 50 minutes for ESF actuation system function 15, boron dilution block. The staff questioned the value used in the safety analysis with

the consideration of instrumentation uncertainties. In its response to RAI-SRP15.0-SRSB-01, the applicant stated that the safety analysis cases reported in DCD Section 15.4.6 support a setpoint of 3.0 over 50 minutes to mitigate the boron dilution event in Modes 3, 4, and 5. It further stated that DCD Table 15.0-4a should also report this value of 3.0 as the setpoint assumed in the safety analysis. The applicant has incorporated this change in DCD, Table 15.0-4a. In its response to RAI-SRP15.4.6-SRSB-02, the applicant stated that the boron dilution analyses performed for Modes 3, 4, and 5 documented in Revision 17 of the DCD, Section 15.4.6, assumed a safety analysis setpoint for the boron dilution protection system of 3.0 over 50 minutes with acceptable results (i.e., the automatic protective actions terminate the boron dilution event and maintain the plant in a subcritical condition). Therefore, the staff concludes that the change in Table 15.0-4a of the DCD, of the limiting setpoint of the boron dilution block on source range flux doubling to 3.0 over 50 minutes, is consistent with the assumption in the safety analysis and, therefore, is acceptable.

Regarding the TS nominal setpoint of 2.2 over 50 minutes, the applicant stated that it has estimated the major factors that contribute to the measurement uncertainty of the boron dilution algorithm. To ensure that the neutron count integrals are as large as possible, the design features include placing the source range neutron detectors outside the reactor vessel along the cardinal axes of the reactor core with the highest neutron leakage, and increasing the counting interval from 60 seconds to 120 seconds in the algorithm. The applicant presumes that the 36 percent difference between the value assumed in the accident analysis and the setpoint selected will bound the value determined when final plant design inputs are available. The staff notes that in DCD Tier 1, Section 2.5.2, "Protection and Safety Monitoring System," Table 2.5.2-8, "Inspections, Tests, Analyses and Acceptance Criteria," design commitment item 10 specifies that the PMS setpoints are determined using a methodology that accounts for loop inaccuracies, response testing, and maintenance or replacement instrumentation. If the measurement uncertainty after the final instrumentation installation is not bounded by the 36-percent allowance, the nominal setpoint would be revised accordingly. Therefore, the staff concludes that there is adequate assurance that the nominal setpoint specified in the TS is consistent with the safety analysis setpoint and the measurement uncertainty.

15.1.0.6.3.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed change of the source range flux doubling setpoint from "1.6 over 50 minutes" to "3.0 over 50 minutes," in DCD Table 15.0-4a is acceptable because this change is consistent with the assumption made in the safety analysis of the boron dilution events with acceptable results.

15.1.0.6.4 Deletion of P-8 Interlock and Replacement with P-10

In Revision 17 of the DCD, the applicant deleted the "High neutron flux, P-8" reactor trip interlock from Table 15.0-4a, and in Table 15.0-6 it changed the permissive interlock for the power range high flux, low-flow trip function credited for the "startup of an inactive reactor coolant pump at incorrect temperature" event from P-8 to P-10.

15.1.0.6.4.1 Evaluation

The power range nuclear power P-8 interlock permits a reactor trip on low flow or reactor coolant pump (RCP) high bearing water temperature in a single loop. In TR-80, the applicant provided the rationale for the deletion of the high neutron flux P-8 interlock from the PMS. As stated in TR-80, Section II.7, "Low Reactor Coolant Flow Reactor Trip Logic Modifications," the

AP1000 is not licensed for N-1 operation and, therefore, the design of the PMS is not required to account for this requirement. The applicant also deleted the P-8 interlock from DCD Table 7.2-3, "Reactor Trip Permissives and Interlocks," and TS Table 3.3.1-1, trip function 10, reactor coolant flow—low, and function 11, RCP bearing water temperature—high. Therefore, the deletion of P-8 from DCD Table 15.0.4a is consistent with the reactor trip logic and TS. The staff concludes that the deletion of the P-8 interlock from DCD Tables 15.04a and 7.2-3 and TS Table 3.3.1-1 is acceptable because the P-8 interlock is not needed for the AP1000 design.

In Revision 15 of the DCD, one of the reactor trip functions used for the "startup of an inactive reactor coolant pump at an incorrect temperature" event is the low-flow trip with P-8 interlock. In Revision 17, the applicant changed the P-8 interlock credited for this event to the P-10 interlock. As discussed above, the applicant deleted the P-8 interlock because the AP1000 is not licensed for N-1 loop operation. The power range nuclear power P-10 interlock, which has a lower setpoint than the P-8 interlock (10 percent and 48 percent power for P-10 and P-8, respectively), performs the same function of allowing reactor trip on low coolant flow or RCP high bearing water temperature in multiple loops that the P-8 interlock performs for single loop. Since the P-10 interlock has lower setpoint than the P-8 interlock, the replacement of the P-8 with the P-10 interlock is conservative because it would permit the reactor to be tripped at a lower power for this event. The staff also requested that the applicant confirm that safety analyses of transient events that take credit for the P-8 interlock have been reanalyzed with the P-10 interlock. The staff agrees with the applicant's response to RAI-SRP15.3.1-SRSB-01 that the replacement of the P-8 with the P-10 interlock will improve the results of the non-LOCA safety analysis, including for loss-of-flow events, at lower power since the reactor will now be tripped by a partial or completed loss of flow down to 10 percent power.

In DCD TS Table 3.3.1-1, the applicant replaced the P-8 interlock with the P-10 interlock for trip function 10, reactor coolant flow—low, and function 11, RCP bearing water temperature—high. The staff concludes that the change to DCD Table 15.0-6 to replace the P-8 interlock with the P-10 interlock is acceptable because it is consistent with the reactor trip logic and the TS.

15.1.0.6.4.2 Conclusion

The staff has reviewed the proposed changes to the DCD for the removal of the power range nuclear power reactor trip P-8 interlock and replacement with the P-10 interlock where applicable. The staff finds these changes conservative and acceptable because the P-10 interlock has a lower setpoint than the P-8 interlock, and the safety analyses of non-LOCA events credited with this interlock would continue to show compliance with GDC 10 and 15 without exceeding the specified acceptable fuel design limits and the RCS pressure boundary design limit.

15.1.0.6.5 Changes in Valve Opening Time Delay

In DCD Table 15.0-4a, the applicant changed the time delay for the automatic depressurization system (ADS) Stage 1 actuation on core makeup tank (CMT) low level signal from 20 seconds to 30 seconds for the control valve to begin to open. It also changed the ADS Stage 4 actuation on CMT low-low level signal from 30 seconds to 2 seconds for the squib valve to begin to open. In Table 15.0-4b, the applicant changed the closure time of the CVS makeup isolation valves from 10 to 30 seconds, and in Table 15.6.5-10, it also changed the control valve actuation and opening times for ADS Stages 1, 2, 3, and 4. It added a note to these tables, stating the following:

The valve stroke times reflect the design basis of the AP1000. The applicable DCD Chapter 15 accidents were evaluated for the design basis stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remains valid. The output provided for the analyses is representative of the transient phenomenon.

15.1.0.6.5.1 Evaluation

In RAI-SRP15.0-SRSB-04, the staff asked the applicant to clarify the relationship among the 2-second delay time of the ADS Stage 4 squib valves specified in Table 15.0-4a and the opening time delay sequences for Substage A and Substage B of ADS Stage 4 described in DCD Section 7.3.1.2.4, "Automatic Depressurization System Actuation," and the ADS Stage 4 actuation time delay assumed in the safety analyses. In its response to RAI-SRP15.0-SRSB-04, the applicant provided the following explanation. In the small-break LOCA analysis in DCD Section 15.6.5.4B, ADS Stage 4 was actuated from the coincidence of CMT level less than the low-2 setpoint, low RCS pressure, and a preset time delay following an open signal initiation for the ADS Stage 3 depressurization valves. The 2-second time delay for ADS Stage 4 actuation on CMT low-2 level identified in Table 15.0-4a represents the accident analysis delay time assumed specifically for the squib valve to begin to open. The accident analysis assumes the 2-second signal processing delay after the CMT low-2 level is reached or the preset time delay after the ADS Stage 3 depressurization valve open signal is generated, whichever is later. This explanation clarifies that the ADS-4 opening time assumed in the safety analysis is for the ADS-4 Substage A depressurization valve opening time, and the Substage B depressurization valve will open after the Stage A valve with a delay time specified in DCD Table 15.6.5-10.

The applicant also identified in its response to RAI-SRP15.0-SRSB-04 the following inconsistencies in Revision 17 of the DCD, Chapters 7 and 15, which need to be revised:

- Section 7.3.1.2.4 states that ADS Stage 2 actuation is interlocked with ADS Stage 1, and ADS Stage 3 actuation is interlocked with ADS Stage 2. However, no such interlocks exist and, therefore, Section 7.3.1.2.4 requires revision.

- Table 15.0-4a indicates an ADS Stage 1 actuation on CMT low-level signal delay time of 30 seconds. This should be revised to 32 seconds, consistent with DCD Table 15.6.5-10, as it includes a 30-second programmable delay time with a 2-second signal proceeding delay.

- DCD Table 15.0.4b incorrectly lists Table 15.6.5-13 for the ADS valve opening times. The ADS valve opening times actually appear in Table 15.6.5-10.

- Table 15.6.5-10 should be modified to add two notes: (1) a description of the interlock of CMT low-2 level, as well as 128 seconds after the ADS Stage 3 actuation signal is generated for initiation of ADS-4 Substage A valves; and (2) that the valve opening time of ADS Stage 4 valves includes an "arm-fire" processing delay.

In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Because the PMS and equipment actuation delay times assumed in the safety analyses for various design-basis events may differ from the valve stroke time design-basis values listed in DCD Tables 15.0-4a, 15.0-4b, and 15.6.5-10, the staff asked the applicant to list the PMS and equipment with the delay times assumed in the Chapter 15 safety analyses that differ from the design-basis values specified in these tables and to evaluate the impact of the inconsistencies in delay time assumptions on the affected events. In its response to RAI-SRP15.0-SRSB-05, the applicant listed the PMS actuation functions and valves with different delay times. The list includes the ADS Stage 1 and Stage 4 actuations on CMT low level signals; the ADS Stage 1, 2, 3, and 4 control (depressurization) valves; and the CVS makeup isolation valves. The events analyzed with different delay times include boron dilution, inadvertent CVS actuation, loss of normal feedwater events analyzed with different CVS makeup isolation valve delay times, and the small-break LOCA and inadvertent operation of the ADS with different ADS valve opening time delays. The applicant evaluated the effects of the inconsistencies on the safety analyses of the affected events.

The design-basis stroke time for the CVS makeup isolation valves is 30 seconds, but the safety analyses for the inadvertent CVS actuation (DCD Section 15.5.2) and loss of normal feedwater (DCD Section 15.2.7) events assume a 12-second valve closure (which includes a 2-second microprocessor delay and 10-second closure time). Based on the CVS makeup flow rate assumed in the analysis, an additional 20 seconds in the makeup valve isolation time would increase the makeup flow into the RCS by about 0.425 cubic meters (m^3) (15 cubic feet (ft^3)). For these two events, there is sufficient available margin between the pressurizer volume and the maximum pressurizer water volume (see DCD Figures 15.5.2-5 and 15.2.7-6, respectively) to accommodate this water volume without changing the conclusion that the minimum DNB ratio remains above the design-limit value and the RCS pressure remains below 110 percent of the design value. For the boron dilution events, the applicant's response to RAI-SRP9.3.6-SRSB-01 provides an evaluation. The longer makeup valve closure time results in no adverse effect for the boron dilution events occurring during Mode 1 and 2 operations because the purge volume of the CVS is not sufficient to return the reactor to criticality. For the events occurring during Mode 3, 4, and 5 operations, the safety analyses described in DCD Sections 15.4.6.2.2 through 15.4.6.2.4 assume a makeup valve closure time of 28 seconds, and there is sufficient margin to accommodate the 2-second delay in the valve closure time.

The only non-LOCA event affected by the ADS is an inadvertent operation of the two ADS Stage 1 trains event, since all other non-LOCA events do not model the ADS valves. The existing analysis described DCD Section 15.6.1 for the inadvertent operation of the ADS Stage 1 valve event assumed a 25-second stroke time, compared to the design-basis value of 40 seconds. As shown in DCD Figure 15.6.1-7, the minimum DNB ratio occurs around 22 seconds, which is before the ADS valves are fully open. Therefore, the longer valve stroke time has no effect on the analysis result and the existing analysis remains valid.

Since the primary role of the ADS is to mitigate a small-break LOCA, the staff asked the applicant to evaluate the effects of changes in the ADS valve stroke times on the small-break LOCA analysis. In response to RAI-SRP5.4.6-SRSB-01, the applicant evaluated the effects of the changes in the ADS Stage 1, 2, and 3 stroke times, as well as the ADS Stage 4 time delay change from 30 seconds to 2 seconds, on the small-break LOCA analysis described in DCD Section 15.6.5.4B, "Small-break LOCA Analyses." The evaluation includes inadvertent ADS actuation, a 2-inch cold leg break, and a direct vessel injection (DVI) break. The results show minimal effects on the safety analysis and that the core remains covered in all cases; thus, the design continues to comply with the acceptance criteria for the ECCS specified in

10 CFR 50.46. Therefore, the staff concludes that the changes in the ADS valve actuation and stroke times specified in DCD Tables 15.0-4a and 15.6.5-10 are acceptable.

It should be noted that in Revision 17 of the DCD, Tier 1, Section 2.1.2, "Reactor Coolant System," and Section 2.3.2, "Chemical and Volume Control System," the applicant proposed to change the acceptance criteria for the opening times after receipt of a signal from the PMS of the remotely operated ADS Stage 1, 2, and 3 valves (RCS-V001A/B, RCS-V002A/B, and RCS-V003A/B) and the CVS makeup isolation valves (V090 and V091) specified in Tables 2.1.2-4 and 2.3.2-4, "Inspections, Tests, Analyses, and Acceptance Criteria," for the RCS and CVS, respectively. The staff finds that the revised values are acceptable because they are consistent with the values described in DCD Tier 2, Tables 15.0-4b and 15.6.5-10.

15.1.0.6.5.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes in DCD Tables 15.0.4a, 15.0.4b, and 15.6.5-10, regarding the delay times of the ADS valves and CVS makeup isolation valves are acceptable because they produce insignificant changes in the results of the analyses for the affected design-basis events, and the acceptance criteria in GDC 10, GDC 15, and 10 CFR 50.46 continue to be met.

15.1.0.6.6 Changes Pertaining to Steam Generator Tube Rupture Analysis

In Revision 17 of the DCD, the applicant made the following changes to Table 15.0-4a:

- Change the high-2 steam generator limiting setpoint from 100 percent to 95 percent of the narrow range level span.

- Add an entry for CMT actuation on pressurizer low-2 water level with a time delay of 2.0 seconds.

15.1.0.6.6.1 Evaluation

In its response to RAI-SRP15.0-SRSB-06, the applicant confirmed that it made these changes to clarify the DCD documentation in areas identified during the AP1000 DCD review process. The changes do not represent new assumptions or results and are consistent with the assumptions credited in the existing Chapter 15 safety analyses. The change of the high-2 steam generator limiting setpoint from 100 percent to 95 percent of the narrow range level span is consistent with the steam generator tube rupture analysis provided in DCD Section 15.6.3. Other events used 100 percent of the narrow range level span, but the steam generator tube rupture analysis provides the limiting setpoint. The addition of the CMT actuation on pressurizer low-2 water level with a time delay of 2 seconds is also consistent with the steam generator tube rupture analysis, which credits this signal. The staff, therefore, concludes that these changes are acceptable.

15.1.0.6.6.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes in DCD Table 15.0-4a pertaining to the steam generator low-2 setpoint and CMT actuation are acceptable because they are made merely for clarification of the DCD documentation and do not represent new assumptions in the steam generator tube rupture analysis.

15.1.0.8 Plant Systems and Components Available for the Mitigation of Accident Effects

15.1.0.8.1 Change to Chemical and Volume Control System Makeup Suction Isolation for Boron Dilution Mitigation

In DCD Table 15.0-6, "Plant Systems and Equipment Available for Transient and Accident Conditions," the applicant changed the equipment credited for the "chemical and volume control system malfunction which results in a boron dilution" event from "low insertion limit annuciators" to "CVS to RCS isolation valves, makeup pump suction isolation valves, from the demineralized water transfer and storage system."

15.1.0.8.1.1 Evaluation

For the event of boron dilution resulting from CVS malfunction, the AP1000 design takes credit for the source range flux doubling signal (as discussed in Section 15.1.0.6.3 above), which will automatically isolate the unborated water source for boron dilution protection. Revision 17 of the DCD changes the CVS makeup valve realignment for boron dilution protection. The applicant revised DCD Section 9.3.6.4.5.1, "Boron Dilution Events," to state that the CVS is designed to address a boron dilution accident by closing redundant safety-related valves, tripping the makeup pumps, or aligning the suction of the makeup pumps to the boric acid tank, or all three. It revised Section 9.3.6.3.7, "Chemical Volume and Control System Valves," to state that these normally open, motor-operated makeup line containment isolation valves close on a source range flux doubling signal to terminate possible unplanned boron dilution events. The applicant also revised Section 7.3.1.2.14, "Boron Dilution Block," to state the following:

> In the event of an excessive increasing rate of source range flux doubling signal, the block of boron dilution is accomplished by closing the chemical and volume control system makeup isolation valves and closing the makeup pump suction valves to the demineralized water storage tanks. This signal also provides a non-safety trip of the makeup pumps. These actions terminate the supply of potentially unborated water to the reactor coolant system as quickly as possible.

Therefore, the staff finds that the change in Table 15.0-6 regarding the use of CVS makeup pump suction isolation valves for the mitigation of boron dilution is consistent with the CVS design. As discussed in Section 15.2.4.6 of this report regarding boron dilution events occurring during Mode 3, 4, and 5 operations, this design change would terminate the boron dilution event sooner and has no safety significant effect on the consequence of the boron dilution events. Therefore, the staff concludes that this change is acceptable.

15.1.0.8.1.2 Conclusion

Based on the above evaluation, the staff concludes that the change in DCD Table 15.0-6, regarding the change to close the makeup line isolation valves for the termination of boron dilution events is acceptable because it reflects the CVS design change, has no significant effect on the consequence of boron dilution events, and the applicable GDC continue to be met.

15.1.0.8.2 Other Changes for Clarification

In Revision 17 of the DCD, the applicant made the following changes to Tables 15.0-6 and 15.0-8:

- Add the main steam isolation valves (MSIVs), startup feedwater isolation, and accumulators credited for the analyses of the inadvertent opening of a steam generator safety valve and steam system pipe failure events (DCD Sections 15.1.4 and 15.1.5).
- Add the steam generator safety valves for the analysis of the inadvertent operation of the CMT during power operation (DCD Section 15.5.1).
- Add the low steamline pressure ESF actuation functions for the analysis of the CVS malfunction that increases reactor coolant inventory (DCD Section 15.5.2).
- Add the low pressurizer level ESF actuation function for the analysis of the steam generator tube rupture (DCD Section 15.6.3).
- Add an entry to specify that the sample line isolation valves are credited for the failure of small lines carrying primary coolant outside containment (DCD Section 15.6.2).
- Revise the footnote to Table 15.0-8 pertaining to the MSIV backup valves from stating "moisture separator reheat steam supply control valve" to "moisture separator reheater 2nd stage steam isolation valves."

15.1.0.8.2.1 Evaluation

In its response to RAI-SRP15.0-SRSB-06, the applicant confirmed that it made these changes merely to clarify the DCD documentation in areas identified during the AP1000 DCD review process. The changes are made to be consistent with the assumptions credited in the corresponding DCD sections in existing Chapter 15 safety analyses, and they do not represent new assumptions or results.

15.1.0.8.2.2 Conclusion

The staff has reviewed the above changes in DCD Tables 15.0-6 and 15.0-8. The staff concludes that these changes are acceptable because they are made merely for clarification of the DCD documentation and do not represent new assumptions or affect the results in the existing Chapter 15 safety analyses.

15.1.0.12 Component Failures

In Revision 17 of the DCD, the applicant proposed to delete the following paragraph in Section 15.0.12.1, pertaining to operator action error:

> A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure. The error is limited to manipulation of safety related equipment and does not include thought process errors that could potentially lead to common cause or multiple errors.

15.1.0.12.1 Evaluation

In Revision 2 of response to RAI-SRP15.0-SRSB-03 dated July 8, 2010, the applicant stated that while the Chapter 15 safety analyses do meet the intent of the statement in DCD Section 15.0.12.1, the statement was originally removed to prevent confusion about the operator action assumptions made for in the DCD safety analyses. There were no changes to the safety analysis assumptions as a result of the removal of this statement. Based on a review of the statement in DCD Section 15.0.12.1, it is determined that the deleted statement will be returned to the DCD, with the punctuation of the statement updated as follows:

> A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure; the error is limited to manipulation of safety related equipment and does not include thought process errors that could potentially lead to common cause or multiple errors.

Therefore, except for the editorial change in the above statement, no change is made to the certified DCD Revision 15 pertaining to operator actions. The staff finds it acceptable.

15.1.0.12.2 Conclusion

As discussed above, the applicant has determined to retain the statement in the certified DCD Revision 15 pertaining to operator actions, except for an editorial change. This is acceptable. In a subsequent revision to the AP1000 DCD, the applicant incorporated this change in the DCD text, which resolves this issue.

15.2 Transients and Accident Analysis

15.2.2 Decrease in Heat Removal by the Secondary System (DCD Tier 2, Section 15.2)

15.2.2.6 Loss of Alternating Current Power to the Plant Auxiliaries (DCD Tier 2, Section 15.2.6)

In Revision 17 of the DCD, Section 15.2.6.2.1, "Method of Analysis," the applicant added the statement that "the main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative." In addition, it deleted the statement from the assumption used in the analysis that "conservative PRHR heat exchanger heat transfer coefficients (low) associated with the low flow rate caused by the reactor coolant pump trip are assumed."

15.2.2.6.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.2.6, "Loss of Power to the Plant Auxiliaries," describes the analysis of the loss of power to the plant auxiliaries caused by a complete loss of offsite grid, accompanied by a turbine-generator trip. Section 15.2.2.6 of NUREG-1793, "Final Safety Evaluation Report

Related to Certification of the AP1000 Standard Design," described the staff's evaluation of this event for compliance with the relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

As stated in Section 15.1.0.3.1 of this report, the confirmation of 1-percent power uncertainty will be performed through COL Item 15.0-1. However, the changes have no effect on the analysis of the loss of alternating current power to the plant auxiliaries because the analysis assumes 2-percent power uncertainty. In addition, the applicant deleted the conservative PRHR heat exchanger heat transfer coefficient to provide more accurate information consistent with the existing analysis of the event. The staff concludes that these changes have no effect on the existing analysis of the event, which continues to show compliance with the relevant acceptance criteria; therefore, the changes are acceptable.

15.2.2.6.2 Conclusion

Based on the above evaluation, the staff concludes that the revisions proposed by the applicant to the DCD, Section 15.2.6.2.1, do not represent new assumptions and have no effects on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.2.7 Loss of Normal Feedwater Flow (DCD Tier 2, Section 15.2.7)

Revision 17 of the DCD added the following statement to Section 15.2.7.2.1, "Method of Analysis," regarding the assumptions used in the analysis: "The main feedwater flow measurement supports a 1 percent power uncertainty; use of a 2-percent power uncertainty is conservative." The applicant also made editorial changes to Table 15.2-1 to correct the time sequence of the loss of normal feedwater flow event; these changes have no effect on the analysis.

15.2.2.7.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.2.7, "Loss of Normal Feedwater Flow," describes the analysis of the loss of normal feedwater flow. Section 15.2.2.7 of NUREG-1793 described the staff evaluation of this event for compliance with the relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

As stated in Section 15.1.0.3.1 of this report, the confirmation of 1-percent power uncertainty will be performed through COL Item 15.0-1. However, the change has no effect on the analysis of the loss of normal feedwater flow because the analysis assumes 2 percent power uncertainty. Therefore, the staff concludes that this change is acceptable.

15.2.2.7.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the DCD, Section 15.2.7, do not represent new assumptions and have no effects on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.2.8 Feedwater System Pipe Break (DCD Tier 2, Section 15.2.8)

Revision 17 of the DCD changes Section 15.2.8.1, "Identification of Causes and Accident Description," to include the high-3 pressurizer water level as a condition for a reactor trip. The applicant also added the statement, "Method of Analysis," that "the main feedwater flow measurement supports a 1 percent power uncertainty; use of a 2 percent power uncertainty is conservative," to Section 15.2.8.2.1.

15.2.2.8.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.2.8, "Feedwater System Pipe Break," describes the analysis of feedwater system pipe break. Section 15.2.2.8 of NUREG-1793 described the staff evaluation of this event for compliance with the relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

In its response to RAI-SRP15.0-SRSB-06, the applicant stated that it added the high-3 pressurizer water level as a reactor trip actuation condition in order to provide more accurate information in the DCD to be consistent with the existing safety analysis assumption. The staff finds the change acceptable since it does not represent a new assumption and has no effect on the analysis result.

As stated in Section 15.1.0.3.1 of this report, the confirmation of 1 percent power uncertainty will be performed through COL Item 15.0-1. However, the change has no effect on the analysis of the feedwater system pipe break because the analysis assumes 2 percent power uncertainty. Therefore, the staff concludes that this change is acceptable.

15.2.2.8.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the DCD, Section 15.2.8, do not represent new assumptions and have no effect on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.3 Decrease in Reactor Coolant System Flow Rate (DCD Tier 2, Section 15.3)

15.2.3.1 Partial Loss of Forced Reactor Coolant Flow (DCD Tier 2, Section 15.3.1)

In DCD, Section 15.3.1, the applicant proposed to change the power range nuclear power reactor trip permissive from the P-8 interlock to the P-10 interlock for protection against the partial loss of forced reactor coolant flow event.

15.2.3.1.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.3.1 describes the safety analysis of the partial loss of forced reactor coolant flow event. Section 15.2.3.1 of NUREG-1793 described the staff evaluation of this event. Therefore, the evaluation in this document addresses the staff evaluation of the proposed change described in Revision 17 of the DCD.

DCD Section 15.3.1.1, "Identification of Causes and Accident Description," describes the cause and progression of the partial loss of forced reactor coolant flow event (i.e., trip of one RCP). The low primary coolant flow reactor trip signal provides protection against this event. Revision 15 of the DCD stated that above permissive P8, low flow in either hot leg actuates a reactor trip, and between approximately 10 percent power (permissive P10) and the power level corresponding to P8, low flow in both hot legs actuates a reactor trip. Revision 17 of the DCD revises this to state that above permissive P10, low flow in either hot leg actuates a reactor trip.

As discussed in Section 15.1.0.6.4 of this report, Revision 17 of the DCD deletes the P-8 interlock because the P-8 interlock, which permits reactor trip on low flow or RCP high bearing temperature in a single loop, is not needed for the AP1000 since it is not licensed for N-1 loop operation. In Revision 17 of the DCD, the applicant deleted the P-8 interlock from Table 7.2-3, "Reactor Trip Permissives and Interlocks." Therefore, the P-8 permissive interlock is changed to P-10 interlock for the low-flow trip function. The P-10 interlock also replaces the P-8 interlock in TS Table 3.3.1-1, trip function 10, reactor coolant flow—low, and function 11, RCP bearing water temperature—high. Therefore, the change to replace the P-8 interlock with the P-10 interlock is consistent with the reactor trip logic and TS. As stated in its response to RAI-SRP15.3.1-SRSB-01, the change from the P-8 interlock to the P-10 interlock will improve the results of the loss-of-flow events at lower powers since the reactor will now be tripped at the P-10 setpoint of 10-percent power, compared to the P-8 setpoint of 48 percent power. As a result of replacing the P-8 permissive with P-10, the reactor trip with low flow in both hot legs when the power level is between approximately 10 percent (permissive P-10) and that corresponding to permissive P8 (48 percent) is not necessary. The staff concludes that the change to DCD Section 15.3.1.1 is acceptable.

15.2.3.1.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the DCD, Section 15.3.1, to replace the P-8 interlock with the P-10 permissive interlock are

conservative relative to the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.3.2 Complete Loss of Forced Reactor Coolant Flow (DCD Tier 2, Section 15.3.2)

In DCD, Section 15.3.2, the applicant proposed to change the power range nuclear power reactor trip permissive from the P-8 interlock to the P-10 interlock for protection against the complete loss of forced reactor coolant flow event.

15.2.3.2.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.3.2, "Complete Loss of Forced Reactor Coolant," describes the safety analysis of the complete loss of forced reactor coolant flow event. Section 15.2.3.2 of NUREG-1793 described the staff evaluation of this event. Therefore, the evaluation in this document addresses the staff evaluation of the proposed change described in Revision 17 of the DCD.

DCD Section 15.3.2.1, "Identification of Causes and Accident Description," describes the cause and progression of the complete loss of forced reactor coolant flow event (i.e., trip of all four RCPs). The low primary coolant flow reactor trip signal provides protection against this event. Revision 15 of the DCD stated that above permissive P8, low flow in either hot leg actuates a reactor trip, and between approximately 10 percent power (permissive P10) and the power level corresponding to P8, low flow in both hot legs actuates a reactor trip. Revision 17 of the DCD revises this to state that above permissive P10, low flow in either hot leg actuates a reactor trip.

Section 15.2.3.1 of this report discusses the staff evaluation that concludes that the replacement of the P8 permissive interlock with the P10 interlock for the low reactor coolant flow is acceptable. This same conclusion applies to the complete loss of forced reactor coolant flow. Therefore, the staff concludes that the change to DCD Section 15.3.2.1 is acceptable.

15.2.3.2.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the DCD, Section 15.3.2, to replace the P-8 interlock with the P-10 permissive interlock are conservative relative to the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.4 Reactivity and Power Distribution Anomalies (DCD Tier 2, Section 15.4)

15.2.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition (DCD Tier 2, Section 15.4.1)

In DCD, Section 15.4.1.1, the applicant proposed to change the high nuclear flux rate reactor trip. This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicates a rate above a preset setpoint. Previously, this trip function could be manually bypassed after the coincident two out of four nuclear power range channels were manually reset. The applicant proposed to no longer allow the manual bypass of the trip function after the manual reset of the coincident two out of four nuclear power range channels.

15.2.4.1.1 Evaluation

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed such that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. Control rod withdrawal is an AOO. The fuel cladding is the first barrier of protection against radioactive release. Meeting GDC 10 ensures that the fuel cladding integrity is not challenged during this AOO.

GDC 13, "Instrumentation and Control," requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to ensure adequate safety, and the provision of controls that can maintain these variables and systems within prescribed operating ranges. Meeting GDC 13 ensures that the appropriate controls are provided to maintain these variables and systems within the prescribed operating ranges.

GDC 17, "Electric Power Systems," requires that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components (SSCs) important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that specified acceptable fuel design limits and design conditions of the RCPS are not exceeded as a result of AOOs. Meeting GDC 17 ensures that the fuel cladding integrity is not challenged during an uncontrolled control rod assembly withdrawal in conjunction with a loss of onsite or offsite power.

GDC 20, "Protection System Functions," requires that the protective system automatically initiate the operation of the reactivity control system to ensure that fuel design limits are not exceeded as a result of AOOs. The withdrawal of a control assembly significantly impacts local fuel pin power and could lead to cladding failure. Measures are required to ensure that an abnormal rod withdrawal is detected and automatically terminated before fuel design safety limits are violated. Meeting GDC 20 ensures that cladding integrity is not challenged during this AOO.

GDC 25, "Protection System Requirements for Reactivity Control Malfunctions," requires that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods. A failure of the reactivity control system that would create an unmitigated withdrawal of a control assembly could lead to cladding failure. Meeting GDC 25 ensures that a power transient fostered from a reactivity addition as a result of a single failure of

the reactivity control system will be detected and terminated before challenging the fuel cladding integrity.

The staff has reviewed the proposed change by the applicant to no longer allow the manual bypass of the high nuclear flux rate reactor trip function after the manual reset of the coincident two out of four nuclear power range channels. Since the proposed change would no longer allow manual bypassing of the trip function, the proposed change has no effect on the analysis results for the control rod withdrawal from subcritical transient. Therefore, the staff concludes that this proposed change is acceptable.

15.2.4.1.2 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design change is acceptable because it continues to meet the requirements of GDC 10, GDC 13, GDC 17, GDC 20, and GDC 25.

15.2.4.3 Rod Cluster Control Assembly Misalignment (DCD Tier 2, Section 15.4.3)

In DCD, Section 15.4.3.1, the applicant proposed to delete a sentence pertaining to a specific operator action upon the inoperability of the rod deviation alarm to be consistent with the TS. The resolution of the rod position indicator channel is ±5 percent of span (±7.5 inches). A deviation of any rod cluster control assembly (RCCA) from its group by twice this distance (10 percent of span or 15 inches) does not cause power distributions greater than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span. The applicant proposed to delete the sentence from the application that states, "If the rod deviation alarm is not operable, the operator takes action as required by the Technical Specification." However, the following paragraph in the application states that if one or more of the rod position indicator channels are out of service, operating instructions are followed to verify the alignment of the nonindicated RCCAs and the operator also takes action as required by the TS.

15.2.4.3.1 Evaluation

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed such that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. Control rod withdrawal is an AOO. The fuel cladding is the first barrier of protection against radioactive release. Meeting GDC 10 ensures that the fuel cladding integrity is not challenged during this AOO.

GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to ensure adequate safety, and the provision of controls that can maintain these variables and systems within prescribed operating ranges. Meeting GDC 13 ensures that the appropriate controls are provided to maintain these variables and systems within the prescribed operating ranges.

GDC 20 requires that the protective system automatically initiate the operation of the reactivity control system to ensure that fuel design limits are not exceeded as a result of AOOs. The withdrawal of a control assembly significantly impacts local fuel pin power and could lead to cladding failure. Measures are required to ensure that an abnormal rod withdrawal is detected and automatically terminated before fuel design safety limits are violated. Meeting GDC 20 ensures that cladding integrity is not challenged during this AOO.

GDC 25 requires that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods. A failure of the reactivity control system that would create an unmitigated withdrawal of a control assembly could lead to cladding failure. Meeting GDC 25 ensures that a power transient fostered from a reactivity addition as a result of a single failure of the reactivity control system will be detected and terminated before challenging the fuel cladding integrity.

In its response to RAI-SRP15.4.3-SRSB-01, the applicant stated that it revised the sentence, "If the rod deviation alarm is not operable, the operator takes action as required by the Technical Specifications," in Revision 16 of the DCD, Section 15.4.3.1, to be consistent with TS 3.1.4 and 3.1.7. The applicant also stated that neither of these TS or any other TS require the rod deviation alarm to be operable. The applicant also added that Revision 3 of the DCD revised TS 3.1.4 and 3.1.7 to be consistent with NUREG-1431, "Standard Technical Specifications—Westinghouse Plants," Revision 2, issued April 2001. NUREG-1431, Revision 2, removes the rod deviation alarm since it serves as indication only and does not directly relate to the limiting conditions for operation. This is documented in TSTF-110, "Delete SR frequencies based on inoperable alarms," Revision 2. AP1000 TS 3.1.4 does require that all shutdown and control rods shall be operable and that individual indicated rod positions shall be within 12 steps of their group step counter demand position. Surveillance Requirement (SR) 3.1.4.1 requires that individual rod positions are verified within alignment limit every 12 hours. The applicant stated that performing this verification every 12 hours provides a history that allows the operator to detect that a rod is beginning to deviate from its expected position. In addition, the specified frequency takes into account other rod position information that is continuously available to the operator in the main control room (MCR) so that during actual rod motion, deviations can immediately be detected. According to the applicant, the digital rod position indication system and the bank demand position indication system make rod position information continuously available to the operator in the MCR. TS 3.1.7 requires the digital rod position indication system and the bank demand position indication system to be operable. The digital rod control system maintains a count of steps taken by each rod group and, based on this information, a digital readout of the demanded bank position is provided and the demanded and measured rod position signals are displayed in the MCR. An alarm is generated whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. The applicant verifies that the alarm is set with appropriate allowance for instrument error and within sufficiently narrow limits to prevent exceeding core design hot channel factors.

The staff has reviewed the proposed change by the applicant to delete the need for operator action if the rod deviation alarm is not operable. Based on its review, the staff finds that this revision does not affect the safety analysis of the RCCA misalignment events described in DCD Section 15.4.3, and the design continues to comply with the relevant requirements. Therefore, the staff concludes that this proposed change is acceptable.

15.2.4.3.2 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design change is acceptable because it continues to meet the requirements of GDC 10, 13, 20, and 25.

15.2.4.6 Chemical and Volume Control System Malfunctions that Result in a Decrease in the Boron Concentration in the Reactor Coolant (DCD Tier 2, Section 15.4.6)

The applicant proposed to revise Section 15.4.6 in the areas related to the CVS design modifications associated with the mitigation of boron dilution events. These modifications include the alignment of the makeup pump suction, and flux doubling boron dilution block and CVS isolation pertaining to boron dilution events occurring during Mode 3, 4, and 5 operations. In addition, the applicant made changes regarding the dilution flow rates, RCS mixing volume, and critical and shutdown boron concentrations for Mode 3, 4, and 5 operations.

15.2.4.6.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs; and GDC 13 and GDC 26, "Reactivity Control System Redundancy and Capability," pertaining to appropriate instrumentation and reactivity control system redundancy and capability.

DCD Section 15.4.6 describes the safety analysis for the boron dilution event. Section 15.2.4.6 of NUREG-1793 described the staff evaluation of this event for compliance with relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

In Section 15.4.6.1, "Identification of Causes and Accident Description," the applicant proposed to modify the description of boron dilution events and the general mitigation method to be consistent with the CVS design changes for boron dilution mitigation described in DCD Section 9.3.6, "Chemical Volume and Control System." The existing design currently realigns the makeup pump suction from the demineralized water tank to the boric acid tank to terminate the potential boron dilution, and to begin to reborate the RCS to restore shutdown margin. These actions would initially cause the boron dilution to continue because the volume of water in the makeup line path would still be unborated until borated water from the boric acid tank began to reach the RCS. The function was changed to close the makeup line isolation valves (as well as the demineralized water isolation valves) or trip the makeup pumps to terminate the event as soon as possible. Long-term recovery from the event would then be accomplished using either a different flowpath with a smaller unpurged volume or by using the makeup line after purging most of the unborated water in it. The applicant also proposed several text changes along with the logic changes that are required to implement this modication.

The staff reviewed the applicant's proposed change related to the realignment from the demineralized water tank to the boric acid tank as well as isolation of the makeup flow to the RCS for the termination of boron dilution events in Modes 3, 4, and 5. Since the proposed change would terminate the boron dilution event sooner and has no safety-significant effect on the transient, the staff finds this change acceptable. The text changes associated with the realignment are, therefore, acceptable.

In DCD, Section 15.4.6.2, the applicant proposed to delete text that states that in the event of an inadvertent boron dilution transient during Mode 3, 4, and 5 operations, the source range nuclear instrumentation detects "an increase of 60 percent of" the neutron flux and replace it

with generic text reading "a sufficiently large increase in" the neutron flux. In its response to RAI-SRP-15.4.6-SRSB-01, the applicant explained that in Revision 15 of the DCD, the use of the phrase "an increase of 60 percent" reflected the 1.6 multiplier (over 50 minutes) setpoint reported in DCD Table 15.0-4a. In Revision 16 of the DCD, and supplemented by TR-80, the applicant proposed to change the multiplier from 1.6 to 2.2 (which would now be an increase of greater than 60 percent). The applicant stated that it reanalyzed the event and verified that an increase in the nominal setpoint from 1.6 to 2.2 demonstrates acceptable results and would reduce the likelihood of an inadvertent actuation of the boron dilution protection functions during normal operation.

The applicant changed Revision 16 of the DCD, Table 15.0-4a, to reflect the proposed 2.2 multiplier. The applicant stated that it decided to remove the text in DCD Section 15.4.6 referencing a specific flux increase value in order to simplify any potential future revisions to the text in the event of a subsequent change in the safety analysis setpoint. By reporting the actual numerical value in DCD Table 15.0-4a, the document completely defines the magnitude of the flux increase modeled in the safety analysis; therefore, the applicant stated that there is no need to repeat that same information in multiple locations within the text. The staff has reviewed the applicant's justification for replacing the text referencing "an increase of 60 percent" with generic text reading "a sufficiently large increase." Since DCD Table 15.0-4a will provide a specific value to quantify the phrase, "a sufficiently large increase," this change is acceptable.

However, in its responses to both RAI-SRP-15.4.6-SRSB-01 and RAI-SRP15.0-SRSB-01, the applicant proposed to increase the multiplier again from 2.2 to 3.0. In its response to RAI-SRP15.0-SRSB-01, the applicant stated that the safety analysis cases reported in DCD Section 15.4.6 support a setpoint of 3.0 over 50 minutes to mitigate the boron dilution event in Modes 3, 4, and 5. However, DCD Section 15.4.6 and TR-80 only discuss the increase from 1.6 to 2.2 and do not mention an increase to 3.0. The applicant submitted Revision 17 of the DCD, Table 15.0-4a, which includes the new proposed flux rate setpoint of 3.0, but the staff asked the applicant to provide additional information regarding the 3.0 multiplier. In response to RAI-SRP15.4.6-SRSB-02, the applicant confirmed that the boron dilution analyses performed for Modes 3, 4, and 5 documented in Revision 17 of the DCD, Section 15.4.6, assumed a safety analysis setpoint for the boron dilution protection system of 3.0 over 50 minutes, and analysis results for all cases are acceptable. The applicant also justified why the 36-percent allowance between the safety analysis and TS nominal setpoints of 3.0 and 2.2 over 50 minutes is sufficient to bound the actual value determined when final plant design inputs are available. The staff reviewed the applicant's response confirming that the boron dilution analyses documented in Revision 17 of the DCD, Section 15.4.6, assumed a safety analysis setpoint for the boron dilution protection system of 3.0 over 50 minutes, and it finds this change acceptable. The staff also reviewed the applicant's justification that the 36-percent allowance between the safety analysis and TS nominal setpoints of 3.0 and 2.2 is sufficient to bound the actual value determined when the final plant design inputs are available, and it finds this explanation acceptable.

Additionally, in DCD, Section 15.4.6.2, the applicant proposed changes to the dilution flowrate, RCS water volume, critical and shutdown boron concentrations, and automatic protective actions initiation time for Mode 3, 4, and 5 operations. The applicant proposed these changes in order to be consistent with the TS, DCD Section 9.3.6, and the assumed conditions for the inadvertent boron dilution event. In its response to RAI-SRP15.4.6-SRSB-03, the applicant provided additional explanation for the changes in the RCS water volumes for different modes of operation. The applicant stated that it recalculated the RCS water volumes using the latest geometric data available, taking into consideration the design changes made up to this point.

These design change refinements result in a change of less than 5 percent for Mode 3 and of less than 8 percent for Mode 5. The Mode 4 RCS volume is the same volume as was assumed in the Mode 3 boron dilution calculation and no credit is taken for the upper head volume in the assumed active mixing volume in the Mode 3, 4, and 5 calculations. The staff reviewed the applicant's rationale for the changes in RCS water volumes for different modes of operation and concludes that these changes are acceptable.

15.2.4.6.2 Conclusion

The staff reviewed the proposed changes described in DCD, Section 15.4.6, regarding the CVS design modifications associated with the mitigation of boron dilution events. Based on the above evaluation, the staff concludes that the AP1000 design changes are acceptable because the analyses of boron dilution events continue to meet the requirements of GDC 10, 13, 15, and 26.

15.2.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents (DCD Tier 2, Section 15.4.8)

DCD Section 15.4.8, "Spectrum of Rod Cluster Control Assembly Ejection Accidents," describes the RCCA ejection events resulting from mechanical failure of control rod mechanism pressure housing. For assemblies initially inserted, the consequences include a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions have been made to render this accident extremely unlikely, the applicant has provided its analysis of the consequences of such an event. The applicant has considered plant systems and equipment, discussed in DCD Tier 2, Section 15.0.8, that are available to mitigate the effects of the event, and it determined that no single active failure in these systems or equipment adversely affects the consequences of the events.

Revision 17 of the DCD removes the longitudinal and circumferential failures described in Sections 15.4.8.1.1.5 and 15.4.8.1.1.6, respectively. These changes are supported by APP-GW-GLE-016, "Impact of In-core Instrumentation Grid, Quicklocs and Changes to Integrated Head Package (IHP)," Revision 0.

15.2.4.8.1 Evaluation

Revision 17 of the DCD removes the failure mechanisms outlined in Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 because of a design change to the upper internals. TR APP-GW-GLE-016 provides a technical description of these changes, but the technical justification for removing the failure mechanisms was unclear.

The staff sent RAI-SRP15.4.8-SRSB-01 to the applicant to ask for more information as to why the failure mechanisms previously covered in DCD Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 are no longer considered. In its response to RAI-SRP15.4.8-01, the applicant stated that it removed the failure mechanisms from Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 because Section 3.9.4.1.1 covers RCCA failure mechanisms, and these methods were no longer considered credible. The RAI response further explains that the hypothetical failure of a RCCA housing described in DCD Section 15.4.8 is a rapid positive reactivity insertion independent of the specific failure mechanism; therefore, DCD Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 are not needed.

The staff reviewed the RAI response and agrees that as long as the analysis continues to evaluate a rapid positive reactivity insertion resulting from an ejected rod, the requirements of

GDC 28, "Reactivity Limits" (as detailed in regulatory guide (RG) 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," issued May 1974, and NUREG-0800 Sections 15.4.8 and 4.2), are met. Therefore, the staff concludes that the proposed change is acceptable.

15.2.4.8.2 Staff Position Related to the Revision of NUREG-0800 Section 15.4.8

In Section 15.2.4.8 of NUREG-1793, the staff provided a safety evaluation of the RCCA ejection analysis in accordance with NUREG-0800 Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Revision 2, issued July 1981. The AP1000 analysis results of the RCCA ejection events initiated at hot full power and at hot zero power demonstrated that the calculated values of the hot spot radially averaged fuel enthalpy are well within the acceptance criterion of 280 calories per gram specified in NUREG-0800 Section 15.4.8, Revision 2.

After the AP1000 design certification (DC) rulemaking, the NRC revised NUREG-0800 Section 15.4.8. Revision 3 of NUREG-0800 Section 15.4.8, issued in March 2007, specifies that the number of failed fuel rods used in the radiological evaluation for the rod ejection events be calculated considering the failure mechanisms addressed in NUREG-0800 Section 4.2. Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," to NUREG-0800 Section 4.2, Revision 3, specifies that, for the reactivity-initiated accidents such as rod ejection accidents in pressurized-water reactors (PWRs), the total number of fuel rods that must be considered in the radiological assessment is equal to the sum of all of the fuel rods failing either: (1) the high cladding temperature failure criteria specified; or (2) the pellet-cladding mechanical interaction cladding failure criteria specified.

In 10 CFR 52.47(a)(9), "Contents of applications; technical information," the NRC specifies that for applications for light-water reactor (LWR) nuclear power plants, the technical information in the application shall include an evaluation of the standard plant design against the NUREG-0800 revision in effect 6 months before the docket date of the application. In addition, 10 CFR 52.63(a)(1), "Finality of standard design certifications," specifies that notwithstanding any provision in 10 CFR 50.109, "Backfitting," while a standard DC rule is in effect under 10 CFR 52.55, "Duration of certification," or 10 CFR 52.61, "Duration of renewal," the Commission may not modify, rescind, or impose new requirements on the certified information, whether on its own motion or in response to a petition from any person, unless the Commission determines in a rulemaking that the change meets one of seven criteria. In 10 CFR 52.79(a)(41), "Contents of applications; technical information in final safety analysis report," the NRC specifies that for applications for LWR nuclear power plant COLs, the technical information in the final safety analysis report (FSAR) shall include an evaluation of the facility against the NUREG-0800 revision in effect 6 months before the docket date of the application. In addition, 10 CFR 52.98(c)(1), "Finality of combined licenses; information requests," specifies that if the COL references a certified design, then changes to or departures from information within the scope of the referenced DC rule are subject to the applicable change processes in that rule. Section VIII, "Processes for Changes and Departures," in Appendix C, "Design Certification Rule for the AP600 Design," and Appendix D, "Design Certification Rule for the AP1000 Design," to 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," specifies the processes for changes and departures from the Tier 1, Tier 2, and Tier 2* information, respectively, in the certified design. For example, Section VIII.B.6 specifies that an applicant or licensee who references this appendix may not depart from Tier 2* information without NRC approval.

Accordingly, the staff determined that the AP600 and AP1000 certified standard designs, which the NRC certified before 2007, are not required to comply with Revision 3 of NUREG-0800 Sections 15.4.8 and 4.2, issued March 2007. For COL applicants or licensees who reference the AP1000 or AP600 certified designs, the staff will review any change or departure from the certified design that requires prior NRC approval as specified in Section VIII of Appendices C and D to 10 CFR Part 52, respectively.

The staff will evaluate the reactivity-initiated accidents such as rod ejection accidents based on the acceptance criteria in effect 6 months before docketing the amendment request, such as the interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, if a change or departure in fuel design or other aspects is proposed that requires a reevaluation of final safety evaluation report Chapter 4, "Reactor," or Chapter 15, "Transient and Accident Analysis."

15.2.4.8.3 Conclusion

The staff concludes that the RCCA ejection analysis meets the acceptance criteria specified in NUREG-0800 Section 15.4.8, Revision 2, which were in effect for the AP1000 DC application. Therefore, the RCCA ejection analysis in the DCD is acceptable and can be used by COL applicants referencing the AP1000 standard design. However, the staff will evaluate the RCCA ejection accident based on the acceptance criteria in effect 6 months before docketing the amendment request, such as the interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, if a change or departure in fuel design or other aspects is proposed that requires a reevaluation of final safety evaluation report Chapters 4 or 15.

15.2.5 Increase in Reactor Coolant System Inventory (DCD Tier 2, Section 15.5)

13.2.5.1 Inadvertent Operation of the Core Makeup Tanks during Power Operation (DCD Tier 2, Section 15.5.1)

The applicant proposed to revise Section 15.5.1.2 by deleting the sentence, "No single active failure in any of these systems or equipment adversely affects the consequences of the accident."

15.2.5.1.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the NUREG-0800. These acceptance criteria include GDC 10 and GDX 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

In Section 15.5.1.2, the applicant proposed to delete the following sentence:

> No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

As stated in DCD Section 15.5.1.2, regarding the PMS actuations for the mitigation of the event, the PRHR heat exchanger removes the core decay heat. The worst single failure assumed is the failure of one of the two parallel isolation valves in the outlet line of the PRHR heat

exchanger to open. Therefore, the single active failure is assumed in the safety analysis. In its response to RAI-SRP15.5.1-SRSB-02, the applicant clarified that it deleted this sentence to remove contradictory information, and the change has no effect on the existing safety analysis assumptions or methodology. Therefore, the staff concludes that this change is acceptable.

15.2.5.1.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to DCD, Section 15.5.1, to delete a contradicting statement are acceptable because there is no effect on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15.

15.2.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (DCD Tier 2, Section 15.5.2)

The applicant proposed to revise Section 15.5.2, "Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory," by deleting the sentence, "No single active failure in any of these systems or equipment adversely affects the consequences of the accident."

15.2.5.2.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

In Section 15.5.2.2, the applicant proposed to delete the following sentence:

> No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

As stated in DCD Section 15.5.2.2, regarding the PMS actuations for the mitigation of the event, the PRHR heat exchanger removes the core decay heat. The worst single failure assumed is the failure of one of the two parallel isolation valves in the outlet line of the PRHR heat exchanger to open. Therefore, the single active failure is assumed in the safety analysis. In response to RAI-SRP15.5.1-SRSB-02, the applicant clarified that it deleted this sentence to remove contradictory information, and the change has no effect on the existing safety analysis assumptions or methodology. Therefore, the staff concludes that the change is acceptable.

15.2.5.2.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to DCD, Section 15.5.2, to delete a contradicting statement to be acceptable because there is no effect on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15.

15.2.6 Decrease in Reactor Coolant System Inventory (DCD Tier 2, Section 15.6, Excluding Section 15.6.5)

15.2.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the Automatic Depressurization System

The applicant proposed to revise Section 15.6.1, "Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS," to change the motor-operated valve stroke times for the ADS valves for Stages 1–3.

15.2.6.1.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

In Section 15.6.1.1, "Identification of Causes and Accident Description," the applicant proposed the following changes to the motor-operated valve stroke times for ADS valves during this event:

- Change the ADS Stage 1 design opening time from 25 seconds to 40 seconds.
- Change the ADS Stage 2 and 3 design opening times from 70 seconds to 100 seconds.

The applicant also proposed to add the following paragraph to clarify the effects of the proposed times above on the analysis results:

> The valve stroke times shown in Chapter 15 tables (input/assumptions) reflect the design basis of the AP1000. The accidents addressed in this section were evaluated for these design basis valve stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remain valid. The output provided in this section for the analyses is representative of the transient phenomenon.

The staff reviewed DCD, Section 15.6.1, and concludes that the proposed changes provide additional clarification and do not have a safety significant effect on the existing safety analysis of the event. Therefore, the proposed changes are acceptable.

15.2.6.1.2 Conclusion

Based on the above evaluation, the staff finds that the addition of the above paragraph in DCD, Section 15.6.1, only provides clarification and does not have a significant effect on the existing safety analysis. Therefore, the staff concludes that the proposed change is acceptable because the relevant requirements of GDC 10 and GDC 15 continue to be met.

15.2.6.5 Loss-of-Coolant Accident (DCD Tier 2, Section 15.6.5)

15.2.6.5.2 Large Breaks (DCD Tier 2, Section 15.6.5.4A)

In Revision 17 of the DCD, the applicant extensively revised Section 15.6.5.4A, "Large-Break LOCA Analysis Methodology and Results." In Revision 15 of the DCD, the applicant performed the best estimate large-break loss-of-coolant accident (BELOCA) analysis using the WCOBRA/TRAC code to calculate thermal-hydraulic transients in the RCS during a postulated large-break LOCA, and it used the HOTSPOT program to calculate the effects of local models on the calculated peak cladding temperature (PCT). The uncertainties associated with the plant input parameters and states, such as initial fluid conditions in the reactor core system and the ECCS boundary conditions, were treated with a response surface method described in Westinghouse Commercial Atomic Power (WCAP)-12945-P-A, "Code Qualification Document [CQD] for Best Estimate LOCA Analysis," Revision 2, issued 1998. In Revision 17 of the DCD, the applicant continued to use the WCOBRA/TRAC and HOTSPOT computer codes for the BELOCA thermal-hydraulic and hot fuel rod analyses. However, it used the Automated Statistical Treatment of Uncertainty Method (ASTRUM) for the statistical treatment of uncertainties, replacing the existing CQD response surface method. The NRC has reviewed and approved ASTRUM, documented in WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method," issued 2005. Consistent with the CQD methodology, ASTRUM follows the steps of the Code Scaling, Applicability, and Uncertainty methodology. ASTRUM differs from the CQD methodology primarily in the statistical technique used for uncertainty treatment. ASTRUM uses a nonparametric statistical technique applied directly to a random sample of outputs, for example, the PCT, the maximum local oxidation (MLO), and the core-wide oxidation (CWO). These sample outputs are computed by applying Monte Carlo sampling of the inputs to the WCOBRA/TRAC and HOTSPOT calculations. The uncertainties and biases remain the same as in the CQD methodology.

In support of the revised BELOCA analysis in Revision 17 of the DCD, Section 15.6.5.4A, the applicant submitted APP-GW-GLE-026, "Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis," Revision 1, issued 2009, which provides a detailed description of the revised analysis.

The staff used NUREG-0800 Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," and RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," issued May 1989, to guide its evaluation of this revised BELOCA analysis.

15.2.6.5.2.1 AP1000 BELOCA Analysis Code Applicability Evaluation

The BELOCA analysis uses the WCOBRA/TRAC code to calculate the effects of initial conditions, power distributions, and global models, and it uses the HOTSPOT code to calculate the effects of local models. The WCOBRA/TRAC code, described in WCAP-12945-P-A, is Westinghouse's best estimate thermal-hydraulic computer code to evaluate the RCS response to a postulated large-break LOCA. Westinghouse developed the code consistent with the guidance provided in RG 1.157 to calculate thermal-hydraulic conditions in the RCS during blowdown and reflood of a LOCA. The code includes the features needed to satisfy the requirements of 10 CFR 50.46(a)(1)(i). The BELOCA analysis of the AP600 standard design used the WCOBRA/TRAC code. Section 21.6.3 of NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," described the staff's review of

the application of WCOBRA/TRAC for the AP600 application. Revision 15 of the DCD used the "2000 Formulation" of WCOBRA/TRAC referenced in WCAP-15644-P, "AP1000 Code Applicability Report," Revision 2, issued 2004. This is the version approved in the analysis of the AP600 site safety analysis report, with the subsequent discretionary and nondiscretionary changes that Westinghouse reported to the NRC in 1998, 1999, and 2000, as presented in Appendix A to WCAP-15644-P, Revision 2. Section 21.6.3 of NUREG-1793 described the staff evaluation of WCOBRA/TRAC applicability to the AP1000 BELOCA.

The approved WCOBRA/TRAC code version used in the ASTRUM evaluation model described in WCAP-16009-P-A is WCOBRA/TRAC Mod 7, Revision 6, which is different from the 2000 Formulation of WCOBRA/TRAC used in Revision 15 of the DCD. The WCAP-16009-P-A code version includes the discretionary and nondiscretionary changes reported through the 10 CFR 50.46 reporting process.

The HOTSPOT program is a one-dimensional conduction code that models a portion of a fuel rod at the PCT or burst location and takes into account fuel relocation following a burst during a large-break LOCA. In the axial node where fuel rod burst is predicted to occur, the fuel relocation model in HOTSPOT is used to account for the likelihood that additional fuel-pellet fragments from above that elevation may settle into the burst region. Sections 25.4.2.3 and 25.4.2.4 of WCAP-12945-P-A describe the HOTSPOT model, application, and assessment. HOTSPOT uses a simple physical model that allows the effects of uncertainties to be calculated directly by running the model with parameter values that vary randomly according to specified distributions. The parameter uncertainties for the local models consider the power rates, fuel and cladding properties, metal-water reaction rates, and heat transfer coefficients.

15.2.6.5.2.1.1 WCOBRA/TRAC and HOTSPOT Code Modifications

Revision 17 of the DCD uses WCOBRA/TRAC (M7AR7_AP) and HOTSPOT (6.1) as the current code versions for the AP1000 revised BELOCA analysis. These versions differ from the earlier versions used in Revision 15 of the DCD and WCAP-16009-P-A because of additional code modifications identified through 10 CFR 50.46 reporting. For completeness, Appendix A to APP-GW-GLE-026 identifies a total of 35 changes through 10 CFR 50.46 reporting since 1998.

Appendix B to WCAP-16009-P-A identified 19 of the 35 changes and formed the bases for the acceptance of WCOBRA/TRAC Mod 7A, Revision 6, for BELOCA analyses. The applicant evaluated each of the changes and concluded that each error and its correction, or discretionary change did not have a significant impact on the WCOBRA/TRAC results and did not affect the prior assessments and uncertainties. The NRC evaluated these in its safety evaluation report for WCAP-16009-P-A. The agency concluded that these corrections were reasonable and effectual and found them to be acceptable. The remaining 16 changes were incorporated into WCOBRA/TRAC (M7AR7_AP) and HOTSPOT 6.1, the current code versions used for AP1000 BELOCA analyses.

Of the 16 changes, 10 are discretionary changes involving either: (1) enhanced input/output, corrections to output edits, or improvement in the automation of running the code cases; or (2) corrections to errors involving an option, model, code, or a code application that was not used for experiment simulations. The applicant evaluated each of these changes and concluded that each discretionary change did not have a significant impact on the WCOBRA/TRAC results and did not affect the prior assessments and uncertainties.

One of the nondiscretionary changes concerned an input error resulting in incomplete solution matrix (see 10 CFR 50.46 letter LTR-NRC-04-17, dated March 25, 2004). Two plant-specific calculations were found to be affected by this error. The applicant confirmed that correction of the error did not change the fundamental LOCA transient characteristics (e.g., blowdown cooling and reflood turnaround timing and behaviors). The reference double-ended guillotine break was used to develop the PCT assessments for each plant. The test simulation affected by this error was also corrected, and the transient calculation repeated. It was found that the error correction had no significant effect on the calculation results, and the prior validation conclusions remain valid. The correction is used for the AP1000 BELOCA analyses.

The second nondiscretionary change concerned Inconel 690 material properties capability (see 10 CFR 50.46 letter LTR-NRC-04-17). The material properties were revised in 2000, but the annual 10 CFR 50.46 report did not include the change. This capability was reported to correct that omission. The Inconel 690 material properties are only used for replacement steam generator analyses where the tube material has changed. The analyses directly reflect the effect of the material properties.

The staff identified the following three nondiscretionary changes with the potential to affect any of the prior code assessment and uncertainty results, and that have an effect on the AP1000 BELOCA analyses:

(1) Revised Blowdown Heatup Uncertainty Distribution

This nondiscretionary change was reported in a Westinghouse 10 CFR 50.46 letter LTR-NRC-05-20, dated April 11, 2005. This error was previously reported in a Westinghouse letter, LTR-NRC-04-11, dated February 3, 2004. As a result of input errors in the loss-of-fluid test (LOFT) facility model used to compare the predicted PCT to the test data to determine this distribution, revised analyses were performed with the version of WCOBRA/TRAC available at that time. As a result of the reanalysis with the modeling error corrections, revised blowdown heatup heat transfer multipliers were developed and the revised cumulative distribution function (CDF) was programmed into the new version of HOTSPOT. The applicant estimated the PCT effect of the revised blowdown heatup CDF by calculating the impact on the reference transient for representative two-, three-, and four-loop plants. The estimates bounded all of the 95th percentile HOTSPOT results. The applicant also made plant-specific estimates of the effect of the revised overall code uncertainty for blowdown for those plants that track the blowdown period.

The applicant identified the errors in the LOFT analyses in its response to RAI-SRP15.6.5-SRSB-03. The most important input error was the flag for the fuel rod gap pressure calculation, which was erroneously set to the steady-state option during the transient calculation. With this flag, the critical heat flux calculation was skipped and the transition to film boiling was only a result of the depletion of liquid from the core region. This was the primary input error that impacted the blowdown heatup heat transfer multiplier distribution. Other modeling aspects of the accumulator and break noding were also updated for consistency with the final, approved version of WCAP-12945-P-A, the one-dimensional pipe to three-dimensional vessel connection input was corrected, and errors in the choked flow flag applied to selected components were corrected.

The development of the blowdown heatup heat transfer multipliers and the resulting revised CDF were consistent with the previously approved methods used in WCAP-12945-P-A. The AP1000 ASTRUM analysis uses this revised CDF for the blowdown heatup heat transfer multipliers, consistent with an ASTRUM analysis of a standard Westinghouse PWR.

(2) Improved Automation of End of Blowdown Time

This discretionary change was reported in 10 CFR 50.46 letters LTR-NRC-05-20 and LTR-NRC-06-8, dated March 16, 2006. The automated selection of the end of blowdown time was first modified by replacing the criterion of 276 kilopascal (kPa) (40 pounds-force per square inch absolute (psia)) with a selection based on the time at which the system pressure stops decreasing. It was again modified by replacing the criterion related to when the system pressure stops decreasing with a selection based on the time when the collapsed liquid level in the lower plenum reaches a minimum and begins to increase again. The redefinition of the end of blowdown is a discretionary change. It has no impact on the WCOBRA/TRAC results and does not affect the prior assessments and uncertainties.

In its response to RAI-SRP15.6.5-SRSB-04, the applicant confirmed that the AP1000 BELOCA ASTRUM analysis used the improved automated method to define the end of blowdown based on the time at which the collapsed liquid level in the lower plenum reaches a minimum and begins to increase again. The time at which the collapsed liquid level is at its absolute minimum is selected as the end of blowdown time. For large double-ended guillotine breaks, similar results are obtained whether the end of blowdown is defined by RCS pressure criteria or at the time the lower plenum collapsed liquid level reaches a minimum. For the smaller breaks sampled in the ASTRUM analyses, the improved definition based on the lower plenum collapsed liquid level is more applicable than the historical pressure criterion. The ASTRUM analyses use a consistent definition of end of blowdown as the time at which the lower plenum collapsed liquid level is at its absolute minimum; this definition is applied for all runs.

(3) HOTSPOT Fuel Relocation

This nondiscretionary change was reported in 10 CFR 50.46 letter LTR-NRC-08-24, dated May 15, 2008. It was discovered that the effect of the fuel relocation on the local linear heat rate was being calculated correctly in accordance with the approved model, but then canceled out later in the coding. The HOTSPOT fuel-relocation error was an error in code logic that does not affect the approval of the fuel-relocation model. The applicant evaluated the impact of this error in letter DCP/NRC2074, "10 CFR 50.46 Report for the AP1000 Standard Plant Design," dated February 15, 2008. The impact was estimated to be Δ 0 °Celsius (C) (Δ 0 °Fahrenheit (F)) during the blowdown phase and Δ 38.9 °C (70 °F) during the reflood phase of the accident. The HOTSPOT fuel relocation error is a nondiscretionary change. It has no impact on the WCOBRA/TRAC results and does not affect the prior assessments and uncertainties. The AP1000 BELOCA analyses use the corrected logic.

The applicant has incorporated all of the 10 CFR 50.46 discretionary and nondiscretionary changes into the current versions of the WCOBRA/TRAC (M7AR7_AP) and HOTSPOT 6.1 codes. The discretionary changes were shown not to have a significant impact on the

WCOBRA/TRAC results and did not affect the prior assessments and uncertainties. The nondiscretionary changes addressed and corrected known errors.

15.2.6.5.2.1.2 WCOBRA/TRAC Code Validation

The phenomena identification ranking table for large-break LOCAs, provided in WCAP-15644-P, Revision 2, and NUREG-1793, indicates that the main difference between the AP1000 and operating PWRs is the DVI. Similar to validation performed for the AP600 design documented in WCAP-14171 and WCAP-14172 (nonproprietary), "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident," Revision 2, additional validation was performed to examine code capability of WCOBRA/TRAC (M7AR7_AP) to model the AP1000 DVI into the downcomer. Appendix B to APP-GW-GLE-026 documents the DVI assessment calculations. Researchers validated the DVI injection by comparing the code calculations with the tests carried out on the Cylindrical Core Test Facility and the Upper Plenum Test Facility.

Calculations for the Cylindrical Core Test Facility Test 78, Run 58, showed a reasonable prediction of the thermal-hydraulic behavior with WCOBRA/TRAC (M7AR7_AP). The filling of the core and downcomer were in reasonable agreement with the data, and the calculated core level was slightly overpredicted and the downcomer level underpredicted. The maximum clad temperatures in the core were also reasonably predicted at most elevations, and they tended to be overpredicted for the higher power rods. Overall, the applicant's evaluation indicated that the WCOBRA/TRAC M7AR7_AP calculations adequately captured the thermal-hydraulic effects associated with downcomer injection, with results similar to those for the AP600.

Calculations for the Upper Plenum Test Facility Test 21, Phases A, B-I, and B-II/III, also showed that the WCOBRA/TRAC (M7AR7_AP) code reasonably and conservatively predicted the downcomer bypass phenomenon, with results similar to those for the AP600.

The applicant's assessments demonstrated that the end-of-bypass remains conservatively calculated. They also demonstrated that the PCT predictions from WCOBRA/TRAC (M7AR7_AP) remain conservative at higher fuel rod elevations and in higher power rods, and essentially best estimate at lower elevations.

Based on its review, the staff concludes that WCOBRA/TRAC (M7AR7_AP) conforms to the guidance provided in RG 1.157 and is acceptable for AP1000 BELOCA analyses to demonstrate compliance with the 10 CFR 50.46 criteria.

15.2.6.5.2.2 WCOBRA/TRAC Nodalization Model for AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Analyses

Appendix C to APP-GW-GLE-026 describes the WCOBRA/TRAC nodalization model for AP1000 BELOCA analyses. The nodalization model was developed using the methodology established in WCAP-12945-P-A. The major portion of a BELOCA analysis involves generating the plant-specific vessel and loop model for the WCOBRA/TRAC analyses. The vessel model, in particular, requires detailed information regarding the reactor pressure vessel internals. The AP1000 nodalization was also compared to the available modeling guidance presented in WCAP-16009-P-A.

15.2.6.5.2.2.1 Reactor Coolant System Loop Model

The AP1000 WCOBRA/TRAC RCS loop model (Figure C-3 in APP-GW-GLE-026) uses 75 components and 87 junctions to model the RCS loops and the passive safety systems, including the vessel module and the interface junctions between the one-dimensional loop components and the three-dimensional vessel module. The model includes the four cold legs, the two hot legs, the two steam generators, and the pressurizer.

The two passive safety injection system loops are modeled separately, complete with the CMT and balance line and the accumulator. The model also includes the in-containment refueling water storage tank (IRWST), the sump connections into the DVI lines, the PRHR, and the ADS.

The ADS modeling is simplified and excludes ADS Stages 1, 2, and 3, located on the top of the pressurizer, because ADS Stage 1 is actuated by a low-level signal from the CMTs, whose flow is effectively shut off by the actuation of accumulators. The CMT liquid level is expected to remains above the ADS Stage 1 actuation setpoint throughout the AP1000 BELOCA cladding temperature excursion, even though CMT injection begins again later in the transient. Therefore, ADS Stages 1, 2, and 3 are not expected to actuate during a large break until long after the PCT is calculated to occur. ADS Stage 4, located on the hot legs, is modeled because the limiting WCOBRA/TRAC case is extended beyond the fuel rod quench time until the CMT liquid level decreases to the low-2 setpoint that actuates the ADS Stage 4 valves and IRWST injection.

The RCS loop model also includes one-dimensional PIPE components, which model the thimble tube bypass in the low-power, support column/open hole, and guide tube assemblies. The inclusion of these components is consistent with the approved modeling of the thimble tube bypass in standard Westinghouse PWRs.

WCAP-16009-P-A describes the cold leg break nodalization used to model the break type (guillotine or split) and size.

It should be noted that Revision 17 of the DCD increased the inside diameter of the pressurizer and decreased the vessel height. However, the internal free volume of the pressurizer and the water volume, at power, remain unchanged (refer to APP-GW-GLR-016) and, therefore, have no effect on the WCOBRA/TRAC loop model. However, the decrease in the pressurizer vessel height results in the changes to the high-3, high-2, and high-1 pressurizer water level setpoints shown in DCD Table 15.0.4a. In addition, in letter DCP/NRC2074, the applicant reported that the pressurizer surge-line resistance used in the DCD Revision 15 WCOBRA/TRAC model was in error. This error has been corrected in this WCOBRA/TRAC model.

15.2.6.5.2.2.2 Reactor Pressure Vessel and Internals Model

For the AP1000 17x17 fuel, each fuel bundle contains 264 fuel rods, 24 thimble tubes, and 1 instrumentation tube. The AP1000 reactor pressure vessel model includes five fuel rod groups, based on the upper internal region design and power levels. Rod 1 represents a single fuel rod, the hot rod, which has the highest power in the core and is located in the hot assembly. Rod 2 represents the remaining 263 fuel rods in the hot assembly. It has a power equivalent to the hot assembly average fuel rods and represents an assembly located beneath a support column. Rod 3 represents the 15,576 fuel rods in the medium-power assemblies located beneath support-column/open-hole channels. Rod 4 represents the 18,216 fuel rods in the

medium-power assemblies located beneath guide-tube channels. Rod 5 represents the 7,392 fuel rods in the peripheral assemblies in the low-power channel.

The location of the vertical section boundaries relative to the reactor pressure vessel structures and internals were chosen consistent with the approved AP600 WCOBRA/TRAC model described in WCAP-14171, Revision 2; NUREG-1512; and WCAP-14601, "AP600 Accident Analyses—Evaluation Models," Revision 2, with the elevations appropriate for the AP1000 design.

The model includes changes to reflect the following design changes to AP1000 reactor internals described in WCAP-16716-NP, "AP1000 Reactor Internals Design Changes," Revision 2, issued May 2007:

- relocation of radial support keys and tapered periphery on the lower core support plate (LCSP)
- addition of a flow skirt in the lower reactor vessel head
- addition of four neutron panels, attached to the outside surface of the core support barrel

The applicant has processes that identify plant configuration changes that could potentially impact safety analyses. It used these internal processes, along with internal processes for assessing evaluation model changes and errors, to identify the need to assess the impacts on LOCA analyses.

As shown in Appendix C to APP-GW-GLE-026, the downcomer region used six azimuthal sectors, as compared to four for the reference model in WCAP-16009-P-A, to account for the DVI lines, similar to the AP600 model. The two additional sectors span two of the four neutron panels, with the remaining two neutron panels falling on gaps (centered between two azimuthal sectors).

The following describes each of the three design changes in detail:

(1) Radial Support Keys/Tapered Periphery on the LCSP

The change to the AP1000 LCSP relocated the four lower radial support keys for the core barrel from the current location of 45 degrees from the cardinal axes to the cardinal locations, which eliminates the potential for interference with the core shroud attachment studs and nuts at the 45-, 135-, 225-, and 315-degree locations. The radial supports keys are now physically aligned with the locations of the cold leg nozzles.

In its response to RAI-SRP15.6.5-SRSB-06, the applicant provided a detailed description of the revised LCSP region modeling for the AP1000 WCOBRA/TRAC model. The metal volume of the LCSP was accounted for in the lower plenum channel. The radial keys were modeled in the downcomer channels to reflect the physical location of the radial keys consistent with the vessel volumes represented by the downcomer channels. One radial key was modeled in each of the two downcomer channel stacks connected to the DVI lines. One half of a radial key was modeled in each of the four downcomer channel stacks connected to the cold legs. The fraction of the radial key in each of these downcomer channels was averaged across the channel because the WCOBRA/TRAC code was not designed to reflect more detailed azimuthal modeling

than that specified by the user through the nodalization and lateral gap connections. The momentum area and loss coefficients were adjusted to calibrate the steady-state pressure drop consistent with the modeling approach used for standard plants and the resolution of the WCOBRA/TRAC model nodalization. The AP1000 ASTRUM steady-state calculation confirmed that the steady-state acceptance criteria, specified in WCAP-16009-P-A, Table 12-6, "Criteria for Acceptable Steady-State," were met. The vessel pressure drop and the vessel inlet nozzle to mid-core pressure drop were benchmarked against hydraulic calculations to be within the acceptance criteria.

(2) Lower Reactor Vessel Head Flow Skirt

One of the design changes to the AP1000 reactor internals involved the addition of a flow skirt in the lower reactor vessel head to obtain a more uniform core inlet flow distribution that meets specifications established by the Westinghouse fuel group. The applicant provided a detailed description of the flow skirt region modeling for the AP1000 WCOBRA/TRAC model in response to RAI-SRP15.6.5-SRSB-07. The metal volume of the flow skirt was accounted for in the lower plenum channel. The radial flow area through the flow skirt holes and between the top of the flow skirt and the bottom of the LCSP was reflected in the gaps that connect the downcomer channels to the lower plenum channel. The AP1000 ASTRUM steady-state calculation confirmed that the steady-state acceptance criteria, specified in WCAP-16009-P-A, Table 12-6, were met. The vessel pressure drop and the vessel inlet nozzle to mid-core pressure drop were benchmarked against hydraulic calculations to be within the acceptance criteria. The applicant determined that the core inlet flow distribution assumptions that formed the basis of the original fuel thermal-hydraulic calculations remain valid.

(3) Neutron Panels

Neutron panels were attached to the outside diameter of the core support barrel to maintain the end-of-life reactor vessel fluence values at less than the maximum allowed in RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, issued May 1988. The panels reduce the reactor vessel fluence at the circumferential locations that have the highest fluence values and provide a relatively rigid structure with 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 (the 0-, 90-, 180-, and 270-degree locations). The DVI lines are located at 0 and 180 degrees, and the hot leg nozzles are located at 90 and 270 degrees. Each neutron panel covers about 30 degrees circumferentially and extends over the entire length of the active core region (4.27 m (14 ft)). The neutron panels are contoured to minimize the impact on the downcomer annulus flow area reduction and to reduce the probability of vortex generation in the downcomer.

The applicant provided a detailed description of the neutron panel modeling for the AP1000 WCOBRA/TRAC model in response to RAI-SRP15.6.5-SRSB-09. The neutron panels were modeled in the downcomer channels to reflect the physical location of the neutron panels consistent with the vessel volumes represented by the downcomer channels. One neutron panel was modeled in each of the two downcomer channel stacks connected to the DVI lines. One half of a neutron panel was modeled in each of the four downcomer channel stacks connected to the cold legs. The fraction of the neutron panel in each of these downcomer channels was averaged across the channel because the WCOBRA/TRAC code was not designed to reflect more detailed azimuthal

modeling than that specified by the user through the nodalization and lateral gap connections. The azimuthal flow areas were modeled as the flow area away from the neutron panels, and the friction factor for azimuthal flow reflects the flow between two walls, consistent with the modeling approach for standard plants. The metal mass of the neutron panels was modeled as an unheated conductor in the appropriate channels. This is an acceptable approach for capturing the effects of the neutron panels in the downcomer, including the calculation of downcomer boiling, and is consistent with the modeling of neutron panels in standard two-loop, three-loop, or four-loop Westinghouse PWRs.

15.2.6.5.2.2.3 Previous WCOBRA/TRAC Modeling Limitation

The AP600 safety evaluation in NUREG-1512, on the acceptability of the AP600 WCOBRA/TRAC model, imposed limitations that require an applicant to address the sensitivity of the CMT and residual heat removal system modeling parameters that are not included in the uncertainty methodology in the event that either the blowdown or reflood phase PCT exceeds 941 °C (1725 °F) for any reason. The applicant addresses this by repeating the study that identifies the PCT sensitivity to CMT/PRHR elimination and adding the blowdown and reflood PCT impacts as a bias to their respective 95 percent PCT results. Section 15.2.6.5.2 of NUREG-1793 also discussed this limitation. Revision 17 of the DCD, Section 15.6.5.4A.5, states that previous AP1000 sensitivity calculations evaluated the sensitivity of the CMT and PRHR to modeling relative to a baseline case. The results show that the calculated PCTs for both the cases, in which the CMT and the PRHR, respectively, were isolated from the rest of the AP1000, were lower than the PCT of the baseline case. In RAI-SRP15.6.5-SRSB-12, the NRC asked the applicant to clarify whether the sensitivity studies for the proposed AP1000 model include the MLO and CWO sensitivities. In its response, the applicant performed sensitivity studies for the CMT and PRHR with the model described in Appendix C to APP-GW-GLE-026. The WCOBRA/TRAC PCT results of the CMT inoperable study showed a temperature decrease of Δ 21 °C (38 °F) when compared to the reference case. The results of the PRHR inoperable study showed a temperature increase of Δ 1.1 °C (Δ 2 °F) when compared to the reference case. However, to perform the PRHR study, the maximum allowable time step had to be reduced by 0.0001 second to execute the case, and a revised reference case was also run with the reduced time step. The revised reference case resulted in a decreased PCT of Δ 3.3 °C (Δ 6 °F) as compared to the original reference case. The applicant concluded that the effect of PRHR inoperability was minimal. Since the AP1000 shows significant margin to the MLO and CWO limits, the staff concludes that no penalties need to be applied to the analysis results for assuming that the safety-related equipment does not operate.

In Attachment 2 to APP-GW-GLE-026, the applicant described the following revisions to Revision 17 of the DCD, Section 15.6.5.4A.5, "Large-Break LOCA Analysis Results," to incorporate the results of these studies:

> The large break LOCA analysis complies with the restrictions in Reference 32 [WCAP-16009-P-A]. AP1000 sensitivity calculations evaluated the sensitivity to modeling of the CMT and PRHR relative to the reference transient configuration. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the reference transient configuration. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was 1 ºC (2 °F) higher than the reference transient configuration.

This is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In summary, the AP1000 ASTRUM BELOCA model was developed consistent with the modeling guidelines presented in WCAP-16009-P-A. Therefore, the staff concludes that the AP1000 ASTRUM BELOCA model using WCOBRA/TRAC (M7AR7_AP) conforms to the guidance provided in RG 1.157 for evaluation models needed to demonstrate compliance with the acceptance criteria in 10 CFR 50.46.

15.2.6.5.2.3 ASTRUM AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Analysis

15.2.6.5.2.3.1 ASTRUM Applicability to AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Evaluation

ASTRUM uses a nonparametric statistical technique applied directly to a random sample of outputs, for example, the PCT, the MLO, and the CWO. These sample outputs are computed by applying Monte Carlo sampling of the inputs to the WCOBRA/TRAC calculations. With this approach, a simple singular statement of uncertainty in the form of a tolerance interval for the numerical acceptance criteria of 10 CFR 50.46 can be developed. Once the desired tolerance level is defined, the number of Monte Carlo code runs required to construct the tolerance interval that meets the desired level of safety can be computed. ASTRUM is based on a 95/95 tolerance level to demonstrate compliance with the 10 CFR 50.46 criteria. This tolerance level requires 124 WCOBRA/TRAC runs.

A confidence interval covers a population parameter with a stated confidence, that is, a certain proportion of the time. A fixed proportion of the population can also be covered with a stated confidence, called a tolerance interval. The endpoints of a tolerance interval are called tolerance limits. An application of tolerance intervals involves comparing specification limits with tolerance limits that cover a specified proportion of the population. The 95/95 statement is interpreted to mean that there is a 95 percent probability that 95 percent of the random output variable population falls within the specified tolerance limits. The maximum and minimum value of the samples of the output variables are used to define the limits. The results obtained with ASTRUM are taken to mean that there is at least a 95 percent confidence that the limiting PCT, MLO, and CWO from the sample exceed the true 95th percentile.

ASTRUM replaced the original CQD response surface method, described in WCAP-12945-P-A, to combine the uncertainties with a direct Monte Carlo sampling method. In the application of CQD methodology to AP1000, the uncertainties in the initial fluid conditions in the RCS and the ECCS boundary conditions were bounded in a direction to maximize the PCT. The model uncertainty component addressed uncertainties in the code models that affected the overall system transient (global models), as well as those that only affected the hot rod (local models). WCOBRA/TRAC was used to calculate the effects of initial conditions, power distributions, and global models, and HOTSPOT was used to calculate the effects of local models. Biases and uncertainties, resulting from the assumption that the initial conditions, the power distribution, and the model uncertainty components were linearly combined, were quantified and taken into account. The CQD methodology calculates the final PCT uncertainty distribution by a combination of response surface equations and Monte Carlo sampling. ASTRUM considers the same plant parameters, but each parameter is randomly sampled for each case. The 95/95 PCT is established using nonparametric order statistics.

With ASTRUM, the number of runs is fixed (124 runs for three outcomes—PCT, MLO, and CWO) and is independent of the number of uncertainty attributes considered in the sampling process. The uncertainty parameters are directly sampled instead of using the bounding approach of the CQD methodology.

ASTRUM retains the distinction between global and local variables. However, in ASTRUM, only a single HOTSPOT calculation is performed for each WCOBRA/TRAC run, instead of the multiple HOTSPOT runs with the CQD methodology used to obtain the local model PCT distribution. The HOTSPOT calculation is now a single calculation where the local uncertainties are set at their values by random sampling from their respective distributions. This is consistent with the Monte Carlo approach, where each uncertainty parameter is randomly sampled from the respective distribution for each simulation, which comprises a WCOBRA/TRAC and a HOTSPOT calculation.

The NRC reviewed the ASTRUM uncertainty methodology and found it to be acceptable for meeting the regulatory requirements of 10 CFR 50.46, as described in the staff evaluation of WCAP-16009-P-A. The AP1000 BELOCA analyses for Revision 17 of the DCD use the previously approved global model uncertainties and biases, as well as the local model uncertainties and biases, including the revised blowdown heatup transfer multiplier. The ASTRUM uncertainty methodology is independent of the physical system being modeled and is equally applicable to the AP1000 BELOCA analyses to demonstrate compliance with the requirements of 10 CFR 50.46.

15.2.6.5.2.3.2 Application of ASTRUM to the AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Evaluation

The ASTRUM uncertainty methodology, used for the AP1000 BELOCA analysis for Revision 17 of the DCD, independently samples the uncertainties of the global models, local models, power distribution, and initial and boundary conditions for each of 124 runs over the same ranges of uncertainty and distributions as in the CQD methodology. The sampled uncertainties become inputs to each of the 124 WCOBRA/TRAC calculations.

The WCOBRA/TRAC thermal-hydraulic boundary conditions for the hot rod are input to the local model uncertainties calculation performed with HOTSPOT. The limiting PCT, MLO, and CWO may come from the same case or as many as three different cases. With ASTRUM, each parameter is assumed to be independent of the other two parameters. This assumption is conservative since the MLO and the CWO depend on the cladding temperature (time at temperature).

The WCOBRA/TRAC studies were performed for the AP1000 BELOCA to determine the sensitivities to some of the major plant parameters. These studies included effects of ranging the steam generator tube plugging, ranging the relative power in the low-power assemblies, loss of offsite power coincident with the break initiation, and the break location. The results were used to identify bounding conditions for these parameters, which were then used in the uncertainty calculations.

APP-GW-GLE-026, Table E-1, "Summary of Plant Physical Description, Initial Conditions, Power Distribution, and Global Model Uncertainty Application in CQD Methodology, AP1000 DCD Methodology, and ASTRUM Methodology as Applied to the AP1000," compares the plant physical models, initial conditions, power distributions, and global model uncertainties

between the CQD method used in Revision 15 of the DCD and ASTRUM used in Revision 17. The values were referenced to the appropriate information in WCAP-16009-P-A.

DCD Table 15.0-4a provided the PMS setpoints and time delay assumed in the accident analyses. As described in Section 15.2.6.5.2.2 of this report, the safety system related to pressurizer water level high-3, high-2, and high-1 setpoints in DCD Table 15.0-4a are consistent with the revised pressurizer dimensions described in APP-GW-GLR-016.

Revision 17 of the DCD, Table 15.6.5-4, provides the major plant parameter assumptions used in the BELOCA analysis. These plant parameters were developed consistent with the guidance provided in WCAP-16009-P-A. The uncertainty distributions for the global models, local models, power-related parameters, and initial and boundary conditions appear in Tables 1, 2, 3, and 4, respectively, of APP-GW-GLE-026. These distributions are consistent with Tables 1-7, 1-8, 1-10, and 1-11, respectively, of WCAP-16009-P-A. The blowdown heatup heat transfer multiplier was modified to correct modeling inconsistencies and input errors in the WCOBRA/TRAC LOFT deck, as described in Section 15.2.6.5.2.1 of this report.

The initial reactor core power of less than 1.01x3400 MWt assumed a calorimetric uncertainty of 1 percent. This is consistent with DCD Table 15.0-2, which lists the initial thermal power output assumed for the large-break LOCA analysis as 3,434 MWt. As stated in the staff safety evaluation in Section 15.1.0.3.1.1 of this report, the applicant will revise DCD Section 15.0.15 to include COL Information Item 15.0.15.1, which requires the COL holder to calculate the primary power calorimetric uncertainty before fuel load to confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated value. Therefore, the staff finds the use of 1 percent initial core power uncertainty acceptable.

The ranges of the parameters were compared to the TS limiting conditions for operation. The accumulator water temperature is ranged based on AP1000 expected values from 10 °C (50 °F) to 48.9 °C (120 °F), which is consistent with the TS 3.6.5, "Containment Air Temperature," limiting condition for operation for an operable containment.

The accumulator pressure range between 4619 kPa (670.0 psia) and 5280 kPa (765.8 psia) was inconsistent with TS 3.5.1, SR 3.5.1.3, which specifies the accumulator pressure range between 4392 kPa and 5302 kPa (637 and 769 pounds-force per square inch gauge (psig)). In addition, Table 15.6.5-4 did not include the accumulator water volume range. In its response to RAI-SRP15.6.5-SRSB-13, the applicant indicated that the incorrect accumulator pressure and water volume ranges were assumed in the AP1000 BELOCA ASTRUM analysis, and a reanalysis of the top 10 HOTSPOT PCT cases from the ASTRUM run set was performed to evaluate the impact of the incorrect pressure and water volume ranges. Maintaining the seed values used in the original ASTRUM analyses, the accumulator pressures and liquid volumes for the top 10 HOTSPOT cases were determined based on the original sampling for each run and the revised ranges. The evaluation showed that the PCT, MLO, and CWO results reported in APP-GW-GLE-026 remained valid for the revised TS ranges. The applicant assessed a 0-degree PCT penalty for this set of closely related errors. It must be noted that Figure 27, "HOTSPOT PCT Versus Effective Break Area Scatter Plot for All 124 Cases," in APP-GW-GLE-026 would likely show some changes in the PCT values for some cases if the complete set of 124 runs were to be reanalyzed with the correct accumulator pressure and water volume ranges. However, the overall conclusions for the limiting break would still be applicable.

The applicant also committed to update Table 15.6.5-4 of the DCD to make the accumulator pressure range and the accumulator water volume ranges consistent with TS 3.5.1, SR 3.5.1.3 and SR 3.5.1.2, respectively. In a subsequent revision to the DCD, the applicant incorporated this change in the DCD.

The percentage of steam generator tube plugging is based on the bounding case (10 percent), and DCD Table 5.1-3, "Thermal-Hydraulic Parameters (Nominal)," provides the nominal RCS thermal-hydraulic parameters for this case. The average RCS temperature (T_{AVE}) listed in Table 15.6.5-4 is consistent with DCD Table 5.1-3 for this level of plugging. The hot assembly location, under a support column, is bounding based on the design of the reactor vessel upper core region internals. The pressurizer location, in the intact loop, is bounding based on AP1000 WCOBRA/TRAC sensitivity calculations.

The single-failure assumption is the failure of one CMT isolation valve to open. During a large-break LOCA, the CMT is actuated by the "S" signal, but the CMT flow is shut off upon accumulator actuation and resumes at the end of accumulator injection. This limiting single failure affects the safety injection flow delivery when the CMT begins to inject toward the end of accumulator injection.

One of the plant parameter assumptions is that the offsite power remains available during the LOCA transient. In its response to RAI-SRP15.6.5-SRSB-14, the applicant provided an offsite power availability/unavailability sensitivity analysis using the AP1000 BELOCA model. It confirmed that the offsite-power-available assumption remains limiting in the AP1000 ASTRUM analysis, because of the effect of RCP operation on the downflow cooling during the blowdown period. With loss of offsite power, the RCPs trip coincident with the break and begin to coast down. With offsite power available, the RCPs continue to run until they are automatically tripped. As a result, more fluid flows from the upper plenum into the hot leg in the broken loop, and less fluid reverses flow from the hot leg of the intact loop into the upper plenum. Therefore, less fluid in the upper plenum is available for blowdown cooling. In the case of a loss of offsite power, the liquid downflow into the hot assembly was observed during blowdown; however, there was no observed downflow of liquid into the hot assembly at the upper core plate with offsite power available. The downflow cooling in the case of a loss of offsite power results in increased blowdown cooling and lower PCT.

The containment backpressure is specified at a bounding minimum value, consistent with the WCAP-16009-P-A methodology. The containment pressure is specified at the break location as an input table. The AP1000 ASTRUM analysis followed the approved WCAP-16009-P-A methodology for determining the conservative containment backpressure approved for standard Westinghouse PWRs. The reference transient was used to establish the containment pressure response that was applied as a boundary condition in the uncertainty analysis calculations. The inputs to the containment pressure calculation were biased to obtain a conservative (low) pressure transient.

The peak linear heat rate, expressed in terms of total heat flux hot channel factor, F_Q, was compared to AP1000 DCD Table 4.3-2, "Nuclear Design Parameters (First Cycle)," and was found to be consistent with the design data. The hot rod assembly power, expressed in terms of nuclear enthalpy rise hot channel factor, $F_{\Delta H}$, was increased from 1.65 to 1.75. In its response to RAI-SRP15.6.5-SRSB-17, the applicant clarified that it increased $F_{\Delta H}$ in the AP1000 ASTRUM analysis in order to provide increased margin for the core design. The hot assembly average integrated power (P_{HA}) was correspondingly increased to 1.683 from 1.586. The hot assembly average integrated power is 4 percent lower than the hot rod integrated power, consistent with

the standard value applied in the approved ASTRUM, WCAP-16009-P-A. DCD Figure 15.6.4A-13 shows the axial power distribution in terms of normalized power integrals in the bottom third of the core and middle third of the core, which is consistent with the treatment in WCAP-16009-P-A.

Only cold leg breaks were analyzed because the hot leg break location was found to be nonlimiting for the BELOCA methodology. ASTRUM explicitly accounts for the effect of break type, and various break types and sizes, such as double-ended cold leg guillotine (DECLG) and split breaks, are assumed to have an equal chance of being sampled. The break size and type were sampled consistent with this methodology. The applicant addressed the limiting break type analysis in its response to RAI-SRP15.6.5-SRSB-14. Before performing the detailed ASTRUM uncertainty analyses, confirmatory calculations were performed to identify the limiting settings for some of the major plant parameters. The results of these calculations were used to define the reference transient case. The applicant found that the important thermal-hydraulic characteristics of the AP1000 during a large-break LOCA were consistent with those observed in conventional Westinghouse three-loop plant analyses. The AP600 large-break LOCA phenomena identification ranking table indicated that the only high-ranking area of difference between the AP600 and a standard three-loop plant with respect to a large-break LOCA was the delivery of emergency core cooling water through the AP600 DVI lines (see the WCAP-14171, Revision 2, and WCAP-15644, Revision 2, WCOBRA/TRAC code applicability reports for the AP600 and the AP1000, respectively). The applicant's experience with BELOCA analyses has shown that, for three-loop plants, either a split break or a double-ended guillotine break may be limiting. The reference transient configuration was determined from simulations of nominal DECLG breaks consistent with the approved ASTRUM for conventional three-loop plants (WCAP-16009-P-A).

For Revision 15 of the DCD, the limiting large-break LOCA analyzed with the CQD methodology was determined to be a DECLG break. In Revision 17, the combination of uncertainty parameters sampled in the ASTRUM analysis resulted in the limiting break being a split break, which is the limiting case for both PCT and MLO. Figure 27 of APP-GW-GLE-026 depicts this result and provides the PCT scatter plot showing the impact of the effective break area on the HOTSPOT PCT analysis.

DCD Section 15.6.5.4A.6 describes the limiting PCT/MLO split break case from the AP1000 ASTRUM analysis with the results shown in Figures 15.6.5.4A-1 through 15.6.5.4A-12. Figure 15.6.5.4A-2 shows that the HOTSPOT PCT occurs during the reflood phase. In its response to RAI-SRP15.6.5-SRSB-14, The applicant confirmed that the AP1000 remains reflood limited and that the design changes did not result in the limiting PCT moving to the blowdown time period. The applicant inspected the 124 ASTRUM calculations. In 100 runs, the PCT occurred during reflood. The applicant then inspected the characteristics of the 24 runs in which the PCT occurred during blowdown. These 24 runs were significantly nonlimiting, with the WCOBRA/TRAC PCTs less than 649 °C (1200 °F). The effective break areas sampled for 21 of these calculations were less than 1.0 times the cold leg area, and one was a split break with an effective break area greater than 2.0. The peaking factor and power shape parameters sampled in the remaining two cases contributed to the low PCTs in these calculations. Overall, the early PCT peak in these 24 runs was attributed to the run-specific combinations of sampled parameters, particularly the sampled break areas for the majority of the runs. The applicant determined that the ASTRUM runs in which the calculated PCT peaks occurred early in the transient resulted from the combination of sampled parameters for those runs and not the AP1000 design changes.

DCD Table 15.6.5-6, "Best-Estimate Large-Break Sequence of Events for the Limiting PCT/MLO Case," provides the time line for the sequence of events for the limiting BELOCA. The containment high-2 pressure (156 kPa (22.7 psia or 8 psig)) is assumed to occur 2.2 seconds after the initiation of the break because the massive size of the break causes an immediate, rapid pressurization of the containment. This assumption of 2.2 seconds is consistent with the time delay for the "S" signal on high-2 containment pressure for a large-break LOCA specified in DCD Table 15.0-4a. In its response to RAI-SRP15.6.5-SRSB-15, the applicant addressed the acceptability of the assumed containment response to the BELOCA spectrum. The 2.2-second assumption overestimates the time to reach the containment pressure high-2 setpoint for a nominal DECLG large-break LOCA. The applicant's calculations with assumptions biased to obtain a conservatively low pressure transient show the containment pressure to be more than 165 kPa (24 psia), as compared to the 156 kPa (22.7 psia or 8 psig)) high-2 setpoint, at 2.2 seconds after the break. While 2.2 seconds may not allow enough time to reach the high-2 containment pressure setpoint for the smallest breaks sampled as part of ASTRUM because of the reduced mass and energy release, these smaller break sizes were found to be nonlimiting for the AP1000 as shown in APP-GW-GLE-026, Figure 27.

DCD Section 15.6.5.4A.5 states that local and core-wide cladding oxidation values have been determined using the methodology approved in WCAP-16009-P-A. The detailed CWO calculation procedure described in Section 11.6-2 of WCAP-16009-P-A would require a series of WCOBRA/TRAC runs based on the oxidation value from the limiting hot assembly rod case, but with varying power levels for the other runs in the series to account for lower power assemblies in other core regions. For the AP1000, the applicant chose to apply the limiting case hot assembly rod oxidation value to the entire reactor core. In its response to RAI-SRP15.6.5-SRSB-18, the applicant clarified that the method used to evaluate the limiting CWO is a conservative approach and in accordance with the procedure used for standard ASTRUM analyses. The results of the 124 cases were ranked by the WCOBRA/TRAC-calculated hot assembly average volumetric oxidation from highest to lowest. The case with the maximum hot assembly average volumetric oxidation was examined. In the AP1000 ASTRUM analysis, this was case "run015" with a hot assembly oxidation of 0.2 percent. Since this oxidation was substantially lower than the regulatory CWO limit of 1 percent, a detailed calculation of the CWO was not necessary for the AP1000. By definition, the CWO fraction is less than the hot assembly average volumetric oxidation because the many lower power assemblies present in the reactor core will have lower average volumetric oxidation than the hot assembly. The application of the limiting hot assembly rod oxidation value to the entire core is conservative and acceptable.

15.2.6.5.2.4 Summary of the Best Estimate Large Break Loss-of-Coolant Accident Analysis Results

The results of the AP1000 BELOCA ASTRUM analyses, based on the 124 WCOBRA/TRAC and HOTSPOT runs, identify the limiting PCT/MLO split break. DCD Table 15.6.5-8 summarizes the AP1000 ASTRUM BELOCA results and shows the calculated 95th percentile PCT to be 1003 °C (1837 °F), the MLO to be 2.25 percent, and the limiting CWO to be 0.2 percent. Table 15.6.5-8 also indicates that the core remains coolable and the core remains cool in the long term. The core coolable geometry is generally satisfied when calculated PCT and MLO are within the associated acceptance criteria. The post-LOCA long-term core cooling, as described in DCD Section 15.6.5.4C, "Post-LOCA Long-Term Cooling," is not affected by the application of ASTRUM and continues to show compliance with 10 CFR 50.46 acceptance

criteria. Therefore, the BELOCA analysis results show a high level of probability that the following criteria of 10 CFR 50.46 will be met:

- The calculated PCT will not exceed 1204 °C (2,200 °F).
- The calculated maximum cladding oxidation will not exceed 17 percent of the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam will not exceed 1 percent of the amount that would be generated if the entire cladding metal surrounding the fuel (excluding the cladding surrounding the plenum volume) were oxidized.
- The calculated changes in core geometry are such that the core remains amenable to cooling.
- After successful initial operation of the ECCS, the core temperature will be maintained at an acceptably low value, and the decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The calculated results for the AP1000 BELOCA meet the acceptance criteria of 10 CFR 50.46 and are, therefore, acceptable.

15.2.6.5.2.5 Conclusion

Based on the above evaluation, the staff concludes that the BELOCA analysis using the WCOBRA/TRAC code and the ASTRUM statistical uncertainty treatment methodology described in Revision 17 of the DCD demonstrates compliance with the acceptance criteria of 10 CFR 50.46 and, therefore, is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

15.2.7 Post-Loss-of-Coolant Accident Long-Term Cooling (DCD Tier 2, Section 15.6.5.4C)

DCD Section 15.6.5.4C did not change; however, the NRC is reviewing the long-term cooling analyses and supporting documentation submitted for resolution of COL Item 6.3.8.2 related to Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sumps," for the AP1000, together with the GSI-191 issues addressed in Section 6.2.1.8, "Adequacy of IRWST and Containment Recirculation Screen Performance." For consistency, the staff's evaluation and conclusion regarding post-LOCA long-term cooling for the AP1000 appear with the GSI-191 discussion in Section 6.2.1.8 of this report.

15.3 Radiological Consequences of Accidents

In DCD Tier 2, Chapter 15, the applicant performed radiological consequence assessments of the following seven design-basis accidents (DBAs), using the hypothetical set of atmospheric dispersion factors (χ/Q values) provided in DCD Tier 1, Table 5.0-1, "Site Parameters"; DCD Tier 2, Table 2-1, "Site Parameters"; DCD Tier 2, Table 15A-5, "Offsite Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis"; and Table 15A-6, "Control Room Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis:

(1) LOCA
(2) fuel-handling accident (FHA)
(3) main steamline break accident outside of containment
(4) RCP shaft seizure accident
(5) failure of small lines carrying primary coolant outside containment
(6) rod cluster assembly ejection accident, and
(7) steam generator tube rupture accident

The applicant concluded in DCD Tier 2, Revision 15, that the AP1000 design will provide reasonable assurance that the radiological consequences resulting from any of the above DBAs will fall within the offsite dose criterion specified in 10 CFR 50.34(a)(1) and the control room operator dose criterion specified in GDC 19, "Control Room," of Appendix A to 10 CFR Part 50. In NUREG-1793, the staff performed independent radiological consequence analyses for all of the DBAs listed above and verified the applicant's assessments.

In September 2008, the applicant submitted Revision 17 of the DCD, as a part of its application to amend the AP1000 DC rule in Appendix D to 10 CFR Part 52. In Revision 17, the applicant requested the following four standard changes to the Chapter 15 of the certified AP1000 DCD, Revision 15:

(1) increase in the assumed decay time in the FHA dose analysis from 24 hours to 48 hours to provide increased radioactive decay of short-lived fission products before the handling of irradiated fuel assemblies

(2) increase in the aerosol removal duration in the containment from 15.5 hours to 24 hours from the initiation of a DBA

(3) revisions to some of the hypothetical offsite and control room χ/Q values, and

(4) changes to the MCR emergency habitability system (VES) operation

The applicant proposed these changes in accordance with the change criterion in 10 CFR 52.63(a)(1)(vii) in that the changes contribute to increased standardization of the certification information.

Subsequent to the submittal of the AP1000 DCD Revision 17, on November 3, 2008, the applicant proposed to add a first-of-a-kind passive control room air filtration line to the MCR VES. This design change is intended to allow the VES to meet the dose acceptance criterion specified in GDC 19 with an allowable control room unfiltered air inleakage of 25.5 cubic meters per hour (m^3/hr) (15 cubic feet per minute (ft^3/min)), which includes 8.5 m^3/hr (5 ft^3/min) for ingress/egress. Section 6.4 of this report provides the staff's evaluation of the MCR VES, including the proposed design changes.

These changes will alter the calculated radiological doses in the control room for the above DBAs analyzed in the certified AP1000 DCD, Revision 15, and will revise some of the χ/Q values listed as site parameters in the following tables:

- DCD Tier 1, Table 5.0-1, "Site Parameters"

- DCD Tier 2, Table 2-1, "Site Parameters"

- DCD Tier 2, Table 15A-5, "Offsite Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis"

- DCD Tier 2, Table 15A-6, "Control Room Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis"

Revision 17 of the DCD, Section 2.3.4, evaluates the proposed changes to the hypothetical short-term accident χ/Qs, and Table 2.3.4-1 lists the accident χ/Q values used as site parameters.

15.3.1 Evaluation

15.3.1.1 Fuel-Handling Accident Decay Time Increase

In Revision 15 of the DCD, Section 15.7.4, "Fuel Handling Accident," the applicant analyzed the radiological consequences of a postulated FHA inside containment and in the fuel-handling area inside the auxiliary building, assuming that a single fuel assembly that has undergone 24 hours of decay time is dropped. This decay time provides for radioactive decay of short-lived fission products to reduce the radiological consequences of a release for the design-basis FHA. Even though the applicant analyzed the FHA using a decay time of 24 hours in Revision 15 of the DCD, the applicant had conservatively specified a 100-hour decay time in Revision 15 of the DCD, Chapter 16, TS 3.9.7, "Decay Time." The longer decay time would result in further reduction of radiological consequences of the FHA than the applicant calculated in the FHA radiological consequences analysis in Revision 15 of the DCD.

As discussed in NUREG-1793, the staff reviewed the applicant's analysis and performed an independent confirmatory radiological analysis for the postulated FHA. The staff's results agree with the applicant's values. Both the applicant's and the staff's results met the relevant dose acceptance criteria at the exclusion area boundary (EAB), low-population zone (LPZ), and control room. Based on its review, the staff finds the applicant's analysis of the FHA to be acceptable and that the dose criteria in 10 CFR 50.34(a)(1) and GDC 19 are met.

In Revision 17 of the DCD, the applicant proposed to change the minimum decay time to 48 hours from the 24 hours assumed in Revision 15 of the DCD and in NUREG-1793. This change will provide additional radioactive decay of short-lived fission products, which would result in a lesser amount of radioactive material available for release and, therefore, lower dose results for the radiological consequence assessment for the postulated FHA. Revision 17 of the DCD, Section 15.7.4, discussed the revised FHA analysis assuming a 48-hour decay time. The staff verified that the only changes to the analyses were related to the change in decay time and the changes in the hypothetical χ/Q values discussed below in Section 15.3.1.3.

The staff finds that the change in the FHA radiological consequence analysis ensures that the decay time assumption in the analysis is consistent with the minimum decay time value proposed for the AP1000 TS and results in lower control room and offsite doses than previously found acceptable for Revision 15 of the DCD. Therefore, the staff finds that the proposed change to the TS decay time is acceptable with respect to the radiological consequences of DBAs. The resulting doses off site and in the control room continue to meet the dose acceptance criteria in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000, and NUREG-0800

Section 15.0.3 for the DC review. The applicant's revised FHA dose analyses show that the dose criteria in 10 CFR 50.34(a)(1), 10 CFR 52.47(a)(2)(iv), and GDC 19 would be met with the change in minimum decay time before fuel movement. The applicant revised the decay time requirement in Revision 17 of the DCD, TS 3.9.7, to 48 hours from 100 hours to be consistent with the revised FHA analysis. The staff finds that the proposed change to the TS is supported by the FHA analysis and, therefore, is acceptable.

15.3.1.2 Aerosol Removal Duration in Containment

In Table 15.6.5-3,"Radiological Consequences of a Loss-of-Coolant Accident with Core Melt," of Revisions 15 and 17 of the DCD, the applicant provided the LPZ doses as 0.238 sievert (Sv) and 0.234 Sv (23.8 and 23.4 roentgen equivalent man (rem)) total effective dose equivalent (TEDE), respectively. In RAI-SRP15.6.5-RSAC-01, the staff asked the applicant to explain the discrepancy in these two LPZ doses, since none of the analysis assumptions given in the certified Revision 15 appear to have changed in Revision 17. In its response, the applicant stated that the analyses in Revision 17 of the DCD took full credit for aerosol removal for the first 24 hours (rather than for 15.5 hours as in Revision 15), using the removal coefficients presented in Appendix 15B to Revision 15 of the DCD. This resulted in a small decrease in the LPZ dose to 0.234 Sv (23.4 rem) TEDE. The staff had previously found the aerosol removal coefficients credited for the first 24 hours in Appendix 15B to Revision 15 of the DCD to be acceptable. Revision 17 did not change any of the aerosol removal coefficients in Appendix 15B. The staff finds that the response to RAI-SRP15.6.5-RSAC-01 is acceptable; therefore, this RAI is closed and the changes to the assumed duration of the aerosol removal in containment are acceptable.

15.3.1.3 Offsite and Control Room χ/Q Values

15.3.1.3.1 Control Room χ/Q Values

In Revision 16 of the DCD, the applicant increased the control room χ/Q values from those specified in Revision 15. In Revision 17, the applicant proposed a change in the control room isolation logic so that the increased χ/Q values in Revision 16 could be maintained, with two exceptions.

The χ/Q values for the heating, ventilation, and air conditioning (HVAC) intake at the ground level containment release points in the 2–8-hour and 8–24-hour intervals are reduced for the LOCA with the nuclear island nonradioactive ventilation system in supplemental filtration mode.

In addition, in Revision 17 of the DCD, the applicant: (1) revised the control room χ/Q values for plant vent/passive containment system air diffuser and ground level containment releases to the control room HVAC intake and Annex Building door; and (2) added new control room χ/Q values for condenser air removal stack releases to the HVAC intake and Annex Building door. Revision 17 of the DCD, Table 2.3.4-1, lists these revisions. Section 2.3.4 of this report provides the staff's evaluation and acceptance of these revisions.

15.3.1.3.2 Offsite χ/Q Values

Revision 17 of the DCD does not change the offsite χ/Q values provided in Revision 15.

15.3.1.4 Offsite and Control Room Doses

15.3.1.4.1 Offsite Doses

The applicant has not changed the radiological doses calculated for the postulated LOCA from those in Revision 15 of the DCD, except for the small decrease for the LPZ dose evaluated in Section 15.3.1.2 of this report. For the radiological consequences of DBAs other than the LOCA, the applicant used the higher and more restrictive χ/Q values provided in Revision 17 of the DCD (compared to those χ/Q values used in Revision 15), while still meeting the dose acceptance criteria. The staff verified and confirmed the applicant's revised offsite doses for the DBAs other than the LOCA. The radiological consequence analyses for the five remaining DBAs (other than the LOCA and FHA as discussed above) have not changed in Revision 17 of the DCD from those described in Revision 15.

Therefore, the DBA analyses in Revision 17 of the DCD used two sets of χ/Q values, one for the LOCA and the other for the DBAs other than the LOCA, although Table 15.A-5 shows only one set of χ/Q values used or LOCA. In its response to RAI-SRP15.6.5-RSAC-02, the applicant stated that Table 15A-5 of a future DCD revision will include these two sets of χ/Q values.

Subsequently, the applicant indicated during the November 3, 2008, meeting with the staff that it will use the same offsite χ/Q values for all DBAs in future DCD revisions to avoid having two sets of χ/Q values, one for the LOCA and another for the DBAs other than the LOCA. In a subsequent revision to the AP1000 DCD, the applicant revised the offsite doses for all DBAs except the LOCA, which resolves this issue.

15.3.1.4.2 Control Room Doses

In Revision 17 of the DCD, the applicant also proposed to modify the control room isolation to include switching to the main control room VES on the pressurizer low-pressure signal. In the LOCA dose analysis, this modification results in actuation of the VES by the time core activity releases start at 10 minutes with leak-before-break approval. In addition, the applicant proposed an effective unfiltered air inleakage assumption of 2.55 m^3/hr (1.5 ft^3/min) into the control room based on a total inleakage of 8.5 m^3/hr (5 ft^3/min), with credit taken for purging of the vestibule door volume and the incomplete mixing of the vestibule and control room volumes with outside air following ingress/egress.

During the November 3, 2008, meeting with the staff, the applicant presented the addition of a new passive control room air filtration line to the VES. This design change is intended to allow the VES to meet the dose acceptance criterion specified in GDC 19 with an allowable unfiltered air inleakage of 15 ft^3/min instead of the assumption of 5 ft^3/min total unfiltered inleakage as discussed above.

To evaluate the revised control room passive filtration design, the staff performed an independent radiological consequence dose calculation for the control room and audited the applicant's dose calculations with the new passive control room air filtration line to the VES. Based on this review of the applicant's analyses, the staff verified that the control room habitability system and the technical support center design in the AP1000 DCD, Revision 17, as modified in RAI response dated May 24, 2010, meet the dose acceptance criteria specified in GDC 19 and NUREG-0800 Section 15.0.3.

The staff previously stated that upon completion of the review and acceptance of this new passive control room air filtration line to the VES, the staff will complete its independent radiological consequence dose calculations for the control room to verify that the control room habitability system and the technical support center (TSC) designed in the AP1000 DCD meet the dose acceptance criteria specified in GDC 19 and NUREG-0800 Section 15.0.3. This was Open Item OI-SRP15.3-1-RSAC-01. With completion of staff's review and an independent dose calculation as discussed above, Open Item OI-SRP15.3-1-RSAC-01 is closed.

Section 6.4 of this report gives the staff's evaluation and acceptance of the new passive control room air filtration line.

15.3.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed DCD changes provide reasonable assurance that the radiological consequences resulting from any of the DBAs will fall within the dose acceptance criteria specified in NUREG-0800 Section 15.0.3, 10 CFR 52.47(a)(2), GDC 19, and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 for the EAB, LPZ, and control room. The staff further concludes that Revision 17 of the DCD also provides reasonable assurance that the AP1000 TSC will meet radiological consequences criteria specified in NUREG-0737, "Clarification of TMI Action Plan Requirements," Supplement 1 and in NUREG-0800 Section 15.0.3.

16. TECHNICAL SPECIFICATIONS

16.1 Introduction

Chapter 16.0, "Technical Specifications," of the AP1000 design control document (DCD), provides the AP1000 generic technical specifications (GTS) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications." Each operating license issued by the Commission is required to include technical specifications (TS) that set forth the safety limits (SLs), limiting safety system settings (LSSSs), limiting conditions for operation (LCOs), and other limitations on facility operation deemed necessary for the protection of public health and safety. 10 CFR 50.36(a)(2) requires, among other things, that each applicant for design certification (DC) include in its application proposed GTS for the portion of the plant that is within the scope of the DC application. For the AP1000 design, these are accepted as documented in 10 CFR Part 52, "License, certifications, and approvals for nuclear power plants," Appendix D, "Design Certification Rule for the AP1000 Design," and NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design."

10 CFR 52.47(a)(11), "Contents of applications; technical information," and 10 CFR 52.79(a)(30), "Contents of applications; technical information in final safety analysis report," states that a DC applicant and a combined license (COL) applicant respectively are to propose TS prepared in accordance with 10 CFR 50.36 and 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors." COL applicants that reference a certified design are to propose plant-specific technical specification (PTS), including the GTS approved during the DC review. The COL applicant may propose deviations from the certified generic TS prior to issuance of the COL by requesting an exemption from the associated 10 CFR Part 52 appendix that codifies the certified design. A holder of a COL may propose changes to the TS in accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," in order to adopt approved changes to the standard technical specification (STS) when such changes apply.

This safety evaluation report (SER) chapter documents the U.S. Nuclear Regulatory Commission (NRC) staff review of the amended GTS proposed by the DC applicant for the AP1000 design and their associated Bases. The review is for completeness and correctness in regard to NRC requirements and guidance, and for consistency with related portions of the DCD. The TS are derived from the analyses and evaluations in the DCD.

16.2 Summary

The AP1000 design employs passive safety-related systems that rely on gravity and natural processes, such as convection, evaporation, and condensation. The AP1000 GTS were modeled after Revision 2 of NUREG-1431, "Standard Technical Specifications: Westinghouse Plants." In some cases, the applicant developed TS beyond those in the STS to account for the advanced passive design features of the AP1000. In many instances, the AP1000 system design functions are similar to those of operating pressurized-water reactors (PWRs), even though the components and systems are new. The amendment to the AP1000 DC affects the following sections of the AP1000 GTS and Bases:

- Section 1
- Section 2
- Sections 3.1 through 3.9
- Section 4
- Section 5

The AP1000 GTS include reviewer's notes stating conditions that a COL applicant (or licensee) must satisfy in order to complete a particular GTS provision (e.g., incorporation of an NRC-approved methodology into a plant's licensing basis, or a staff determination that a licensee's probabilistic risk assessment (PRA) program is of adequate quality).

In some instances, detailed design information, equipment selection, instrumentation settings, or other information needed to establish appropriate TS and Bases was not provided during the review of the AP1000 DC or the amendment to the AP1000 DC. This information is identified in Chapter 16 of the DCD and in the GTS and Bases and will be included in the PTS by the applicant for a COL. Locations for the addition of this information are signified in the GTS by square brackets [ ] to indicate that the COL applicant must provide plant-specific values or alternative text.

As parts of the amendment to the AP1000 DC, the applicant proposed to complete some of the bracketed COL information items. Technical report (TR)-74A (APP-GW-GLR-064), "AP1000 Generic Technical Specifications Completion," Revisions 0 and 1, were submitted to document these changes.

The remaining changes to the AP1000 GTS are either results of modifications to the plant equipment designs or are to resolve inconsistency between various TS requirements and their supporting information in the associated TS Bases. Revisions 0 and 1 to TR-74C (APP-GW-GLN-075), "AP1000 Generic Technical Specifications for Design Changes," were submitted to document these changes.

The applicant also submitted TR-134 (APP-GW-GLR-134), "AP1000 DCD Impacts to Support COLA Standardization," to document any supplemental changes to the AP1000 GTS that were not included in TR-74A or TR-74C.

This SER addresses changes to the AP1000 DCD, since Revision 15. These revisions were prepared using the guidance in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," and to the extent applicable for the DC, using regulatory guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants," as a guide for format and content.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): There are no ITAAC for this area of review.

TS: TS are provided in DCD Tier 2, Chapter 16.

16.3 Regulatory Basis

16.3.1 Regulatory Requirements

The relevant requirements of the NRC's regulations for this area of review, and the associated acceptance criteria, are given in Chapter 16 of NUREG-0800, and are summarized below. Review interfaces with other NUREG-0800 sections can be found in Chapter 16 of NUREG-0800.

Section 182a of the Atomic Energy Act of 1954 (AEA), as amended, requires that applicants for nuclear power plant operating licenses will state:

> such technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the NRC established its regulatory requirements related to the content of TS. In doing so, the NRC placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences. As recorded in the Statements of Consideration, 33 *Federal Register* (*FR)* 18610, "Technical Specifications for Facility Licenses; Safety Analysis Reports," (December 17, 1968), the NRC noted that applicants were expected to incorporate into their TS "...those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Accordingly, 10 CFR 50.36(c) requires that TS include: (1) SLs and LSSSs, (2) LCOs, (3) surveillance requirement (SRs), (4) design features, and (5) administrative controls.

10 CFR 50.36(c)(2)(ii) requires that an LCO be established in TS for each item meeting one or more of the following four criteria:

- Criterion 1 - Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary (RCPB).

- Criterion 2 - A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 3 - A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 4 - An SSC shown by operating experience or a probabilistic safety assessment to be significant to public health and safety.

In accordance with 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 17, "Electric Power Systems"; GDC 21, "Protection System Reliability and Testability"; GDC 34, "Residual Heat Removal"; GDC 35, "Emergency Core Cooling"; GDC 38, "Containment Heat Removal"; GDC 41, "Containment Atmosphere Cleanup"; and GDC 44, "Cooling Water," those SSCs shown to be significant to public health and safety need to have sufficient independence, redundancy, and testability to perform their safety functions assuming single failure.

10 CFR 50.36a, requires that TS include procedures for control of radioactive effluents.

10 CFR 52.47(a)(11) requires that a DC applicant propose TS prepared in accordance with 10 CFR 50.36 and 10 CFR 50.36a.

16.3.2 Regulatory Guidance

The relevant NRC requirements for TS and Bases reviews, and the associated acceptance criteria, are given in Chapter 16 of NUREG-0800. They are summarized below. Areas of review that interface with other NUREG-0800 sections can also be found in Chapter 16 of NUREG-0800.

For the reasons discussed in detail below, the acceptance criteria adequate to meet the above requirements are included in the STS documents. The STS for PWRs are in three NRC NUREGs. For each NUREG, Volume 1 includes the TS and Volume 2 includes the associated TS bases. The STS include bases for SLs, LSSSs, LCOs, and associated action and SRs. The NUREGs for the STS for PWRs are as follows:

- NUREG-1430, "Standard Technical Specifications Babcock and Wilcox Plants"
- NUREG-1431, "Standard Technical Specifications Westinghouse Plants"
- NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants"

The STS reflect the results of a detailed review of the application of the Interim Policy Statement criteria to generic system functions, which were published in a "Split Report" issued to the nuclear steam supply system vendor owners groups in May 1988. The STS also reflect the results of extensive discussions concerning various drafts of STS so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all reactor designs. As such, the generic Bases presented in the NUREGs provide an abundance of information regarding the extent to which the STS present the requirements necessary for protecting public health and safety.

On July 22, 1993, the NRC issued its Final Policy Statement (58 FR 39132), expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the AEA and 10 CFR 50.36. In the final policy statement, the NRC described the safety benefits of the STS and encouraged licensees, to the extent applicable to use the STS for PTS amendments and for complete conversions to improved TS. Major revisions to the STS were published in 1995 (Revision 1), 2001 (Revision 2), and 2004 (Revision 3).

The format and content for GTS and Bases prepared for a DC should use STS and applicable Bases to the extent possible, notwithstanding design-specific characteristics. As is appropriate, deviation from the STS, as well as design-specific characteristics, should be technically justified by an applicant and reviewed in detail by the NRC prior to approval.

16.3.3 Other Guidance

The June 2005 "Writer's Guide for Plant-Specific Improved Technical Specifications," prepared by the Technical Specifications Task Force (TSTF), provides specific guidance for the preparation of PTS. The purpose of the guide is to provide guidance on the format and content of the improved TS and to promote consistency in content, format, and style.

Design/plant-specific risk insights were developed by the staff for use during the review of AP1000 applications and are provided in a risk insights report. The risk insights were developed using information from the AP1000 DCD and AP1000 PRA. The risk insights were used to identify areas that warranted more detailed review and to identify equipment and systems that met Criterion 4 in 10 CFR 50.36(c)(2)(ii).

16.3.4 Applicable Generic Communication

The following generic communications issued by the NRC are TS-related and require consideration when developing TS and associated bases:

- Generic letter (GL) 88-016, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 3, 1988
- GL 91-004, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," April 2, 1991
- GL 96-003, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996
- GL 03-001, "Control Room Habitability," June 12, 2003
- GL 06-001, "Steam Generator Tube Integrity and Associated Technical Specifications," January 20, 2006

The following NRC generic safety issues (GSI) are TS-related and require consideration when developing TS and associated Bases:

- GSI-78, "Monitoring of Fatigue Transient Limits for Reactor Coolant System"
- GSI-120, "On-Line Testability of Protection Systems"

16.4 Evaluation

The staff reviewed and evaluated the GTS and Bases to verify their accuracy and completeness. The staff also reviewed the GTS to confirm the appropriateness of the restrictions imposed by the GTS to ensure that an operating AP1000 will operate within its SLs and LSSSs, as described in the final safety analysis report (FSAR). The GTS must ensure that a plant designed and constructed in accordance with the AP1000 design will be operated so as to maintain the validity of the analyses in the FSAR during the operating lifetime of the plant. In particular, the GTS must require an AP1000 licensee to take specified actions, up to and including shutting down the plant, if one or more SSCs are not functioning as designed, such that the plant may not respond as predicted in the FSAR, including the accident analyses in

FSAR Tier 2, Chapter 15. In addition, the GTS must include provisions to govern every SSC that meets one or more of the four criteria in 10 CFR 50.36(c)(2)(ii).

As described in more detail below, the staff verified the adequacy of the GTS primarily by comparing them with the STS developed for the operating fleet of power reactors. The staff developed each of these sets of STS by generically applying the criteria of 10 CFR 50.36(c)(2)(ii) to the SSCs included in the respective designs. Whether any set of STS is adequate to govern the operation of a particular power reactor cannot be determined without an evaluation of the TS as applied to the SSCs of the particular plant, considering the design as a whole. Currently, 75 of the 104 units of the operating fleet of nuclear plants use the STS, in whole or in part; the majority of these units use the Westinghouse STS in NUREG-1431.

While the staff has not approved the STS on a generic basis, it has implicitly approved them on a case-by-case basis through staff review of license amendment requests in which licensees of currently operating reactors have proposed to incorporate STS provisions in the existing custom technical specification (CTS) in their operating licenses. Some amendments have involved adoption of applicable STS on an item-by-item basis, while others have involved entire conversions of a plant's CTS to improved TS incorporating most, if not all, of the STS applicable to the particular design involved. The staff has evaluated and confirmed the adequacy of the model STS to ensure that particular plant SSCs will be operated in accordance with the analyses in individual plant FSARs in the context of these amendment requests. In addition, licensees of currently existing plants have employed STS pursuant to amendment requests granted by the NRC to govern the operation of their plants, and the staff has not identified any adverse effect on plant safety due to the adoption of the STS. Accordingly, the STS can be used as a model for the GTS to govern the operation of SSCs to the extent the AP1000 SSCs are similar in design and function to those governed by the STS. Use of the STS as guidance for the evaluation of the GTS in this manner allows the staff to determine whether the operation of SSCs in accordance with the GTS will assure that the analyses in the FSAR for these SSCs remain valid during plant operation. In view of its 2 Loop PWR design and the similar functions of many of its SSCs to the SSCs of a Westinghouse 4 Loop design, the Westinghouse STS in NUREG-1431 can be applied as guidance in evaluating most of the AP1000 GTS.

The staff evaluated each of the changes in the respective TS sections listed below. The applicant committed to making the changes in the final version of the AP1000 DCD that are identified in the AP1000 DCD, Revision 17. In a subsequent revision (Revision 18) of the AP1000 DCD, the applicant included these changes. In the review of confirmatory items in Revision 18, some conforming changes to the TS were identified. In addition, a number of editorial corrections were found. This resulted in several DCD changes for Chapter 16 that are included in Revision 19.

The staff did not review sections of the AP1000 GTS and Bases that were unaffected by the changes proposed in the AP1000 DCD. The technical evaluation for the sections that were not affected by the amendment can be found in NUREG-1793.

16.4.1 Use and Application

Section 1.0 of the AP1000 GTS includes definitions of terms used in the context of plant TS, and examples to illustrate the applications of logical connectors, completion times for required actions, and frequencies for SRs. Changes to AP1000 GTS Section 1.0 are as follows:

- In TS Section 1.1, the applicant proposed changes to the definition of "SHUTDOWN MARGIN," which is used in conjunction with TS Sections 3.1.1, 3.1.4, 3.1.5 and 3.1.6, to clarify how the gray rod cluster assemblies (GRCA) will be accounted for in the calculation of SHUTDOWN MARGIN. In request for additional information (RAI)-SRP16-CTSB-01, the staff requested additional details regarding this change. In its response, dated November 11, 2008, the applicant provided the requested information including a markup of changes to TS Section 1.1 in AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. Section 16.4.3.1 below provides the staff's evaluation of proposed changes to TS Section 3.1.

- In addition, the staff noted that an error in TS Section 1.4 had not been corrected in accordance with the NRC approved TSTF-485, "Correct Example 1.4-1," which corrects Example 1.4-1, Revision 0. RAI-SRP16-CTSB-02 was issued to the applicant for its correction. In its response, dated December 2, 2008, the applicant agreed to revise TS Section 1.4 in a future DCD revision. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the definitions for terms, logical connectors, conventions for completion times, and frequency requirements as provided in the Westinghouse STS. In addition, AP1000 GTS Section 1 and its Bases do not include any "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 1.0 of the AP1000 GTS is acceptable.

16.4.2 Safety Limits

Section 2.0 of the AP1000 DCD GTS and Bases include requirements for SLs, to ensure that the fuel design limits are not exceeded during steady state conditions, normal operational transients and anticipated operational occurrences.

The specifications provided in Section 2.0, which include the reactor core SLs and the reactor coolant system (RCS) pressure SL, are consistent with the STS and are found acceptable by the staff. Changes to AP1000 GTS Section 2.0 are as follows:

- In RAI-SRP16-CTSB-66, the staff asked the applicant to make an editorial change regarding an acronym in the bases of Section 2.1.1. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable section in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the SL information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 2 and its Bases do not include any "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 2.0 of the AP1000 GTS and Section B 2.0 of the AP1000 Bases are acceptable.

16.4.3 Limiting Condition for Operation and Surveillance Requirement Applicability

Section 3.0 of the AP1000 GTS and Bases includes general provisions regarding determination of equipment operability and performance of SRs in specific TS Section 3-series (i.e., TS Sections 3.1 through 3.9). There is no proposed change to AP1000 GTS Section 3.0.

16.4.3.1 Reactivity Control Systems

Section 3.1 of the AP1000 GTS and Bases includes requirements for the reactivity control systems, which are designed to reliably control reactivity changes and ensure that the capability to cool the core is maintained under postulated accident conditions.

The specifications provided in Section 3.1 consists of: Sections 3.1.1, "Shutdown Margin"; 3.1.2, "Core Reactivity"; 3.1.3, "Moderator Temperature Coefficient"; 3.1.4, "Rod Group Alignment Limits"; 3.1.5, "Shutdown Bank Insertion Limits"; 3.1.6, "Control Bank Insertion Limits"; 3.1.7, "Rod Position Indication"; 3.1.8, "Physics Tests Exceptions – Mode 2"; and 3.1.9, "Chemical and Volume Control System Demineralized Water Isolation Valves and Makeup Line Isolation Valves," are consistent with the STS and are found acceptable by the staff. Changes to AP1000 GTS Section 3.1 are as follows:

- In RAI-SRP16-CTSB-34, the staff asked the applicant to clarify the mode of applicability for an SR in the bases of Section 3.1.1. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable section in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-67, the staff asked the applicant to make a minor editorial change regarding the title of an LCO in the Bases portions of Sections 3.1.4 and 3.1.8. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-05, the staff asked the applicant to clarify certain notes and their corresponding applicability modes in the specification and Bases portions of Sections 3.1.4, 3.1.5, and 3.1.6. In a letter dated November 19, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16CTSB-60, the staff asked the applicant to make an editorial change regarding required actions stated in the Bases of Section 3.1.7. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-43, the staff asked the applicant to make an editorial change regarding the required reactor power level stated in the specification and Bases portions of Section 3.1.8. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-20, the staff asked the applicant to make an editorial change regarding the correct revision year for a reference used in the Bases portion of Section 3.1.8. In a letter dated December 9, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the reactivity control systems information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.1 and its Bases do not include any "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.1 of the AP1000 GTS and Section B 3.1 of the AP1000 Bases are acceptable.

16.4.3.2 Power Distribution Limits

Section 3.2 of the AP1000 GTS and Bases includes requirements for the reactor core power distribution limits, which are designed to reliably control core thermal limits and core power distribution consistent with the design safety analysis. Changes to AP1000 GTS Section 3.2 are described as follows:

- The specifications provided in Section 3.2, which consists of Sections 3.2.1, "Heat Flux Hot Channel Factor"; 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor"; 3.2.3, "Axial Flux Difference"; 3.2.4, "Quadrant Power Tilt Ratio"; and 3.2.5, "OPDMS-Monitored Parameters," are consistent with the STS and are found acceptable by the staff.

- In RAI-SRP16-CTSB-68, the staff asked the applicant to make an editorial change regarding the documentation of the use of a reference in the Bases portion of Section 3.2.3. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-23, the staff asked the applicant to clarify the mode of applicability stated in the specification and Bases portions of Section 3.2.5. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the power distribution limits information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.2 and its Bases do not include any "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.2 of the AP1000 GTS and Section B 3.2 of the AP1000 Bases are acceptable.

16.4.3.3 Instrumentation

Section 3.3 of the AP1000 GTS and Bases include requirements for the instrumentation systems that display information required to protect against violating the core fuel design limits and RCS, and to mitigate accidents. Changes to AP1000 GTS Section 3.3 are described as follows:

- Section 3.3, "Instrumentation," of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 16, and TR-134, Revisions 0 through 5, the applicant made minor editorial changes and updated technical information. The applicant justified other editorial changes in TR-74C, Revision 0. RAI-SRP16-CTSB-69 was submitted to correct editorial errors in these changes. In the AP1000 DCD, Revision 17, the applicant corrected the editorial and typographical errors.

- The applicant removed the brackets [ ] around the completion times in Sections 3.3.1 and 3.3.2., and restored the 92-day frequency to SR 3.3.1.6 and SR 3.3.2.5. The applicant documented the basis for these changes in TR-74, APP-GW-GLR-064, Revision 1, April 13, 2007, "AP1000 Generic Technical Specifications Completion Update on Open Items," and APP-GW-GSC-020, Revision 0, March 17, 2008, "AP1000 Protection and Safety Monitoring System Technical Specification Completion Time and Surveillance Frequency Justification." The applicant has incorporated these changes in the AP1000 DCD, Revision 17. The applicant revised the SR completion times to be consistent with APP-GW-GSC-020.

- The applicant stated that ALL values specified for Trip Setpoints and Allowable Values, in Tables 3.3.1-1 and 3.3.2-1, must be confirmed following the completion of the plant-specific setpoint study. After selection of specific instrumentation, the Trip Setpoints can be calculated using the setpoint methodology described in Westinghouse Commercial Atomic Power (WCAP)-16361, APP-PMS-JEP-001, Revision 0, May 2006, "Westinghouse Setpoint Methodology for Protection Systems – AP1000." In the AP1000 DCD, Revision 17, the applicant has removed all bracketed items for Trip Setpoints and Allowable Values in the tables, but includes a Reviewer's Note to direct the COL applicant to use the approved methodology to calculate these values. A discussion of the acceptability of the AP1000 Setpoint Control Program (SCP), used to calculate setpoint values with this methodology, is included in Section 16.4.5 of this report.

- In TS 3.3.1, Table 3.3.1-1, equations for overtemperature ΔT (Note 1) and overpower ΔT (Note 2) are provided. The staff previously requested, in RAI-SRP16-CTSB-42, that the applicant provide the technical bases and derivation of the revised overtemperature ΔT and overpower ΔT reactor trip setpoint equations presented in Revision 16, and provide a reference to a document approved by the staff for the basis of the revised equations, or submit the basis for the revised equations to the staff for further review. The response provided for RAI-SRP16-CTSB-42 did not fully address the staff's request. WCAP-8745-P-A, previously reviewed and approved by the staff, provided the bases for the overtemperature ΔT and overpower ΔT setpoint equations presented in Revision 15 of the DCD. The revised equations presented in the DCD Revision 16 for these reactor trip functions differ from those previously submitted in Revision 15 of DCD Section 7.2.1.1.3 and TS Table 3.3.1-1, Note 1.

  Based on this, the staff believed that the applicant should document the bases for the revised equations; the bases for development of the tables of allowable core thermal power as a function of core inlet temperature at various pressures for the overtemperature ΔT trip equation, the bases for the determination of the preset bias K4 in the overpower ΔT trip equation, and the bases for the constants and bracketed values that appear in the revised equations presented in Revision 16. The staff reviewed TR

APP-GW-GLR-137, Revision 0, "Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OTΔT) Reactor Trips," submitted by the applicant, in WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Over Temperature ΔT Trip Functions," September 1986 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML073521507). The content of this review is found in Chapter 7, Appendix 7.A of this report. The applicant described and clarified these items in more detail, as well as provided commitments to update references in the report and add this additional information. The staff finds these acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-44, the staff requested clarification/consistency of Function 6 (overtemperature ΔT) and Function 7 (overpower ΔT) "required channel" column in Table 3.3-1, "Reactor Trip System Instrumentation." The applicant added "4 (2/loop)" in the required channel column for clarification. This has been reviewed and accepted by the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-45, the staff requested clarification/consistency of Function 12 (reactor coolant pump [RCP] speed-low) "required channel" column in Table 3.3-1, "Reactor Trip System Instrumentation." The applicant added "4 (1/pump)" in the required channel column for clarification. This has been reviewed and accepted by the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-52, the staff requested resolution of conflicting information for the required minimum number of core exit thermocouples per core quadrant. The conflict was between Note (b) in Table 3.3.3-1, "Post Accident Monitoring," and DCD Table 7.5-1, "Instrumentation and Controls," Sheet 2. The applicant changed the number of instruments required from "2 quadrants" to "2 quadrants per Division" in Table 7.5-1, Sheet 2. This has been reviewed and accepted by the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the instrumentation information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.3 and its Bases include "bracketed information." The staff reviewed each piece of "bracketed information" to understand its intent and to determine whether each was site-specific and appropriately deferred to applicants for construction permits or COLs that reference the AP1000 GTS. The staff concluded that each such item was indeed plant- or site-specific. Therefore, the staff finds that Section 3.3 of the AP1000 GTS and Section B 3.3 of the AP1000 Bases are acceptable.

16.4.3.4 Reactor Coolant System

Section 3.4 of the AP1000 GTS and Bases include requirements for various RCS parameters (i.e., pressure, temperature, flow, etc.) and subsystems (i.e., RCS loops, pressurizer, low-temperature overpressure protection (LTOP), etc.) to ensure the fuel integrity and the RCPB integrity are preserved during all modes of plant operation. Changes to AP1000 GTS Section 3.4 are described as follows:

- In TS 3.4.1, the applicant proposed to use the preliminary bracketed value of 1.41×10^6 Liters per minute (Lpm) (301,670 gallons per minute (gpm)), specified in LCO 3.4.1.c for the minimum RCS total flow rate, as a final value based on latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. The staff finds this final value acceptable since it is consistent with supporting information provided in the TS Bases B 3.4.1 and relevant information described in AP1000 DCD Sections 4.4 (Table 4.4-1) and 15.0 (Table 15.0-3).

- The applicant also proposed to change requirements specified in SR 3.4.1.4 for monitoring RCS flow, to reflect an alternate testing method to the precision heat balance (an NRC-accepted method). In RAI-SRP16-CTSB-25, the staff asked the applicant to provide justification for the change. In its December 2, 2008, response, the applicant provided additional details about the basis for the alternate method and also stated the following:

  > The intent of the proposed Section 3.4.1 is to permit either method, whichever is demonstrated to provide less measurement uncertainty....The total uncertainty in measuring flow will depend upon analysis of the baseline flow measurements and the accuracy of the devices used to periodically measure dP caused by RCS flow. If the total uncertainty is not shown to be less than for the precision heat balance plus Delta-T method, then the alternate method would not be used.

  The applicant also indicated that no change to the AP1000 DCD or the TS 3.4.1 and associated bases is required.

  In reviewing this response, the staff noted that the alternate testing method using elbow tabs had been approved for use at the South Texas Project Electric Generating Station. A review of the current South Texas Project TS found the following descriptions for the affected SRs:

  > SR 4.2.5.2 The RCS flow rate indicators shall be subjected to a channel calibration at least once per 18 months.
  >
  > SR 4.2.5.3 The RCS total flow rate shall be determined by precision heat balance or elbow tab dP measurements at least once per 18 months.

  Based on the above, the staff believed a revision to the SR 3.4.1.4 and TS Bases 3.4.1 was needed to incorporate additional details regarding the choice of a testing method that produces better uncertainty analysis results, including a new SR for a channel calibration of the RCS flow rate indicators. In its response, dated August 20, 2009, the applicant proposed to: (1) add a new SR for a channel calibration for RCS flow indicators at the main control room board; and (2) revise SR 3.4.1.4 and TS Bases B 3.4.1 to incorporate a discussion of uncertainty analyses related to the use of elbow tabs as an alternate method for RCS flow verification. The staff finds this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 3.4.2, the applicant proposed to use the preliminary bracketed value of 288 °Celsius (C) (551 °Fahrenheit (F)), for the minimum RCS cold-leg temperature for criticality, as a final value based on historical relationships between the no-load

operating temperature (292 °C (557 °F)), the minimum temperature for criticality (288 °C (551 °F)), and the limit for Mode 2 physics testing (283 °C (541 °F)). The staff finds this final value acceptable since it is consistent with supporting information provided in the TS Bases B 3.4.2 and relevant information described in AP1000 DCD, Sections 5.4 and 15.0.3.

- In TS 3.4.4, the applicant proposed to replace the preliminary bracketed values of 135 °C (275 °F) with a new final value of 93 °C (200 °F) and to use the preliminary bracketed value of 10 °C (50 °F) as a final value, regarding temperature requirements for the primary coolant and the secondary-side water as listed in Note 2 of LCO 3.4.4. In addition, the applicant proposed to add an extra precautionary note regarding restrictive plant conditions before starting an RCP for the reactor vessel LTOP. In RAI-SRP16-CTSB-55, the applicant was asked to provide clarification of the selected value of 93 °C (200 °F). This value 93 °C (200 °F) is not consistent with the one listed in the Westinghouse STS 135 °C (275 °F). In the March 23, 2009, response letter, the applicant proposed a further change from 93.3 °C ("200 degree F") to 177 °C ("350 degree F") based on an updated LTOP analysis, which now credits a technical design difference for AP1000 related to the variable-speed RCP start-up design limitations (e.g., RCPs are required to be started at a relatively slow pump speed and they are unable to start at full speed) not recognized in the Westinghouse STS (NUREG-1431). The staff finds the changes and justification as provided in this latest response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- Although no change was proposed to TS 3.4.6 as part of the AP1000 DC amendment application, the staff noted inconsistencies between SR 3.4.6.1 requirements and supporting information in Bases B 3.4.6, regarding lift setpoints for pressurizer safety valves. In RAI-SRP16-CTSB-08, the staff asked the applicant to address these inconsistencies. In its response dated December 17, 2008, the applicant proposed to revise the Bases for SR 3.4.6.1 to indicate +/- 1 percent OPERABLE range for the valve lift settings, to be consistent with SR 3.4.6.1 and with the tolerance established in WCAP-16779, "AP1000 Overpressure Protection Report," April 2007. The staff finds this change acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 3.4.7, the applicant proposed to delete LCO 3.4.7.d for the total primary-to-secondary leakage through both steam generators (SGs) because it is redundant to LCO 3.4.7.e for primary to secondary leakage through any one SG. In addition, changes were proposed to SR 3.4.7.2 to reflect the implementation of a new SG program to maintain the SG tube integrity. The staff agreed with the applicant's position on deleting LCO 3.4.7.d and finds the proposed changes to SR 3.4.7.2 acceptable since they are consistent with other requirements in the AP1000 GTS (GTS 3.4.18, GTS 5.5.4, and GTS 5.6.8).

- In TS 3.4.8, the applicant proposed to add a missing clarifier to the applicability statement that allows stopping all RCPs without having to enter an action statement. The staff finds this change acceptable since the added special plant condition is consistent with remaining TS 3.4.8 requirements.

- The applicant also proposed to replace the preliminary bracketed value of 37,785 Lpm (10,000 gpm) for the minimum RCS flow with a final value of 11,356 Lpm (3,000 gpm). Conforming changes were proposed in SR 3.4.8.1 and related information in the TS Bases to match the new minimum flow value (e.g., the minimum pump speed setting of 25 percent was replaced with a new value of 10 percent). The applicant cited the NRC-accepted response to RAI 440.106 during the Revision 14 AP1000 DC review as justification for the proposed flow reduction. In RAI-SRP16-CTSB-62, the staff asked the applicant to provide additional details to support these changes. In its response dated December 17, 2008, the applicant reiterated information that was provided in the response to RAI 440.106 but also stated the following:

  > AP1000 RCS flow calculations show that the expected RCS flow with a single reactor coolant pump (RCP) operating at its lowest allowable operating speed is approximately 64,352 Lpm (17,000 gpm). The associated reactor vessel flow is approximately 41,640 Lpm (11,000 gpm). This is well above the 11,356 Lpm (3,000 gpm) flow mixing requirement from the LOFT testing, and also above the preliminary bracketed value of 37,854 Lpm (10,000 gpm).

  The staff noted that the new proposed value of 10 percent for the pump minimum speed setting in SR 3.4.8.1, corresponding approximately to a calculated flow of 29,810 Lpm (7,875 gpm), appears to be inconsistent with the lowest allowable operating speed stated above. In its response, dated July 15, 2009, the applicant stated that the mentioned lowest allowable operating speed is for equipment protection while the minimum speed specified in TS 3.4.8 is to reflect the assumed RCS flow through the core in shutdown event analyses. The TS limit will be satisfied if the operating limit is maintained in accordance with plant equipment operating procedures. The staff finds this response acceptable and RAI-SRP16-CTSB-62 is closed.

- Although no change was proposed to TS 3.4.11/12 as part of the AP1000 DC amendment application, the staff noted that the scope of Condition A was not clearly defined. In RAI-SRP16-CTSB-07, the staff asked the applicant to explain the difference in scope of inoperable equipment involved between TS 3.4.11/3.4.12 Condition A, which states "One required flow path inoperable," and Condition B, which states, "One required stage 1 ADS flow path inoperable AND Either one required stage 2 or stage 3 ADS flow path inoperable."

  In its October 27, 2008, response letter, the applicant stated the following:

  > As described in the 3.4.11 and 3.4.12 Bases, Conditions A and B cover two different combinations of ADS flow path inoperabilities.. Separate Conditions are specified, since both Conditions A and B may be entered at the same time. The inoperabilities covered by the two Conditions are permissible at the same time, since the safety function can be accomplished by the remaining seven ADS flow paths without a single failure. The loss of capacity while in Conditions A and B is equivalent to a single failure of the power to the valves in one division, as considered in the accident analyses.

  The applicant further stated "the LCO 3.4.11 and LCO 3.4.12 and associated Bases are technically correct, as-is. However, to clarify the system status while in both

Conditions A and B the following statement is added in each of the Bases at the beginning of the Actions sections:

> The loss of automatic depressurization system (ADS) capacity, if both Conditions A and B are entered at the same time, is equivalent to a single failure of the power to the valves in one division, as considered in the accident analyses."

Based on this response and considering the four-stage ADS design, the staff believed that additional changes were required for Condition A to explicitly list Stage 4 ADS flow path in its scope and to clearly indicate the difference between Conditions A and B. In its July 15, 2009, response, the applicant proposed to revise all Action statements in TS 3.4.11 and TS 3.4.12, and the associated TS bases to clearly define the scope of inoperable ADS valves for each LCO condition and to assign a completion time consistent with guidance in the STS for cases with the same remaining operable ADS valves. The staff finds this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 3.4.14, the applicant proposed to replace the preliminary bracketed value of 152.4 square centimeters (cm^2) (9.3 square inches (in^2)) for the minimum RCS vent area with a final value of 68 cm^2 (4.15 in^2). In RAI-SRP16-CTSB-35, the staff asked the applicant to provide a justification for the change. In its response dated December 12, 2008, the applicant stated that the change is a result of the final design of the normal residual heat removal system (RNS) suction relief valve with its inlet changed from 10.16 centimeters (cm) (4 inches(in)) to 7.62 cm (3 in). The staff finds the stated justification acceptable since either the RNS suction relief valve or a depressurized RCS with a vent area is considered an acceptable means for providing LTOP. The staff considers RAI-SRP16-CTSB-35 closed.

- In addition, in RAI-SRP16-CTSB-54, the staff asked the applicant to address inconsistencies in the TS bases B 3.4.14 regarding discussion on restarting of one RCP as a heat input event. In its March 23, 2009, response, the applicant proposed to revise LCO 3.4.14 and its associated TS bases to make the descriptions of LCO 3.4.14, Notes 1 and 2 consistent with those specified in LCO 3.4.4 and LCO 3.4.8 for the same situation. The staff finds this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 3.4.15, the applicant proposed to use the preliminary bracketed value of 15,272 kilopascal (kPa) (2,215 pounds per square inch gauge (psig)) and 15,549 kPa (2,255 psig), for the range of RCS pressure during performance of SR 3.4.15.1 to verify leakage through each RCS pressure isolation valve, as final values based on the nominal RCS pressure design of AP1000 and the requirements for test pressures identified in American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code ISTC-3630(b). The staff finds the stated reason for the final selected values acceptable.

- At the end of TS Section 3.4, the applicant proposed to add a new TS 3.4.18, "Steam Generator Tube Integrity," to reflect implementation of the NRC-approved TSTF-449,

"Steam Generator Tube Integrity," Revision 3. The staff finds the proposed addition of TS 3.4.18 acceptable since implementing TSTF-449 is one acceptable option for addressing the safety issues identified in GL 06-001.

The applicant adhered to the RCS information as provided in the Westinghouse STS, with differences to reflect AP1000 unique design features. With respect to AP1000 unique design features, the GTS are sufficient to assure operation of these features within the bounds of the safety analysis. In addition, AP1000 GTS Section 3.4 and its Bases do not include any "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.4 of the AP1000 GTS and Section B 3.4 of the AP1000 Bases are acceptable.

16.4.3.5 Emergency Core Cooling Systems

Section 3.5 of the PTS and Bases includes requirements for the safety-related passive core cooling system (PXS), which is designed to perform emergency core decay heat removal, RCS emergency makeup and boration, and safety injection. Changes to AP1000 GTS Section 3.5 are described as follows:

- In TS Section 3.5.2, the applicant proposed to use the preliminary bracketed value of 0.0057 cubic meters (m^3) (0.2 cubic feet (ft^3)), for the maximum allowable volume of noncondensable gases in each of the core makeup tanks' inlet piping, as a final value based on latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. The staff finds this final value acceptable since it is consistent with related information described in AP1000 DCD Section 6.3.

- In TS Section 3.5.4, the applicant proposed to replace the preliminary bracketed value of 0.011 m^3 (0.4 ft^3), for the maximum allowable volume of noncondensable gases in the passive residual heat removal heat exchanger inlet piping, with a new final value of 0.025 m^3 (0.9 ft^3). In RAI-SRP16-CTSB-36, the staff asked the applicant to provide justification for the change. In its response dated December 12, 2008, the applicant stated that the value of 0.025 m^3 (0.9 ft^3) reflects the correct design value based on the final location for the alarm limit switch installed in the high-point pipe stub section. The staff finds this final value acceptable based on verification that a physical change was made in AP1000 DCD Section 6.3, regarding an increase in pipe size at the level switch location from 0.305 meters (m) to 0.355 m (12 in to 14 in). Therefore, the staff considers RAI-SRP16-CTSB-36 closed.

- In TS Sections 3.5.6 and 3.5.8, the applicant proposed to replace the preliminary bracketed value of 2091 m^3 (73,900 ft^3), for the minimum volume of borated water in the in-containment refueling water storage tank (IRWST), with a new final value of 2069 m^3 (73,100 ft^3). In RAI-SRP16-CTSB-37, the staff asked the applicant to provide justification for the change. In its response dated December 12, 2008, the applicant stated the following:

  > The bracketed volume of 2091 m^3 (73,900 ft^3) represented a preliminary estimate of the minimum design basis IRWST water volume.
  >
  > The un-bracketed value of 2069 m^3 (73,100 ft^3) was updated based on evolving IRWST design details, is consistent with the updated IRWST volume provided in DCD Table 6.3-2 (Sheet 2), and reflects a more

conservative water volume that was appropriately used in safety analyses."

The staff finds the stated reason acceptable and considers RAI-SRP16-CTSB-37 closed.

The applicant adhered to the ECCS information as provided in the Westinghouse STS. In addition, the AP1000 GTS, Section 3.5, and its Bases do not include "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.5 of the AP1000 GTS and Section B 3.5 of the AP1000 Bases are acceptable.

16.4.3.6 Containment Systems

Section 3.6 of the AP1000 DCD GTS and its Bases address requirements for the containment systems, which are designed to contain fission products that may exist in the containment atmosphere following accident conditions.

The specifications provided in Section 3.6 consist of: Sections 3.6.1, "Containment"; 3.6.2, "Containment Air Locks"; 3.6.3, "Containment Isolation Valves"; 3.6.4, "Containment Pressure"; 3.6.5, "Containment Air Temperature"; 3.6.6, "Passive Containment Cooling System (PCS) – Operating"; 3.6.7, "PCS Shutdown"; 3.6.8, "Containment Penetrations"; and 3.6.9, "pH Adjustment," are consistent with the STS and are found acceptable by the staff. Changes to AP1000 GTS Section 3.6 are as follows:

- In RAI-SRP16-CTSB-15, the staff asked the applicant to correct Bases B 3.6.6 to accurately reflect the action statements in TS 3.6.6. In a letter dated October 17, 2008, the applicant acknowledged the need for the change and included a markup of the applicable sections in AP1000 DCD, Revision 17. The staff finds this change acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-16, the staff asked the applicant to correct Bases B 3.6.7 to accurately reflect the action statements in TS 3.6.7. In a letter dated October 17, 2008, the applicant acknowledged the need for the change and included a markup of the applicable sections in AP1000 DCD, Revision 17. The staff finds this change acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-13, the staff asked the applicant to clarify Bases B 3.6.4 regarding maximum peak containment pressure. In a letter dated December 2, 2008, the applicant acknowledged the need for clarification and included a markup of the changes that will be incorporated. The staff finds this change acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16.1.1-SEB1-01, the staff asked the applicant to provide details on the equipment hatch and bolt design to ensure that the equipment hatch can be safely installed with four bolts to meet the containment closure requirements during Modes 5 and 6 (TS 3.6.8). In its response dated August 15, 2008, the applicant stated that design specification document APP-MV50-Z0-002, "Equipment Hatch Design

Certification Document," would provide final design information for the equipment hatch installation.

In an audit on November 30, 2009, the staff confirmed that design specification document APP-MV50-Z0-002, Revision 1, provides design criteria for equipment hatch bolts, such that its weight can be supported with only four bolts installed. A typical vendor report prepared for the AP1000 China Sanmen Unit 1 lower equipment hatch by IHI Corporation in Japan was also provided for the staff's review during this audit. The staff noted that this vendor report adequately addresses the applicable design criteria specified in APP-MV50-Z0-002. Although this vendor report was not prepared specifically for a new AP1000 plant to be built in the United States, the staff finds the applicant's approach to address this issue acceptable. The equipment hatch is listed as an ASME Code Section III component in AP1000 DCD Tier 1, Table 2.2.1-1, and as such, ITAAC Item 2.a in Table 2.2.1-3 is applicable to it. This ITAAC item calls for the existence of a design report for the as-built component received from the equipment supplier. The design document APP-MV50-Z0-002, together with the cited ITAAC item, will ensure equipment hatch components will be constructed and installed in accordance with design requirements. Therefore, the staff finds the response acceptable and RAI-SRP16.1.1-SEB1-01 is closed.

- In RAI-SRP16-CTSB-61, the staff asked the applicant to specify the sections of DCD Chapter 15 that support the specific accident discussed in the "Applicable Safety Analyses" section of TS Bases B 3.6.1, B 3.6.2, and B 3.6.3. In its November 19, 2008, response, the applicant stated that:

  > the level of detail provided by the B 3.6.1, B 3.6.2, and B 3.6.3 Bases references to Chapter 15 is consistent with that provided in the STS

  The applicant made no further change to the Bases. The staff found this reason unacceptable. The staff's concern was that DCD Chapter 15 is voluminous as it includes more than 600 pages. Without references to specific sections, validation of the information discussed in the affected TS bases would require significant effort and time from the plant operators who implement TS requirements and often refer to the TS bases for clarifications needed quickly. In its July 15, 2009, response, the applicant provided sufficient details to support its position that reference to specific sections of Chapter 15 is not helpful in these cases for the containment boundary as a physical barrier. The containment integrity and leak tightness are applicable to a wide range of accident scenarios described in Chapter 15. The staff finds this response acceptable and RAI-SRP16-CTSB-61 is closed.

- In RAI-SRP16-CTSB-33, the staff asked the applicant to provide the value of the minimum tri-sodium phosphate (TSP) manufactured density that is used to convert the required TSP amount from a mass number to a volume number. In its December 12, 2008, response, the applicant did not provide the requested information so that the staff could verify the accuracy and completeness of supporting information provided in TS Bases B 3.6.9. In its August 20, 2009, response, and the subsequent October 29, 2009, conference call, the applicant provided the requested information and agreed to revise TS Bases B 3.6.9 to include the new details. The staff finds this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the containment information as provided in the Westinghouse STS, with differences to reflect AP1000 unique design features. With respect to AP1000 unique design features, the GTS are sufficient to assure operation of these features within the bounds of the safety analysis. In addition, AP1000 GTS Section 3.6 and its Bases include "bracketed information" and "Reviewer's Notes." The staff reviewed each piece of "bracketed information" to understand its intent and to determine whether each was site-specific and appropriately deferred to applicants for construction permits or COLs that reference the AP1000 GTS. The staff concluded that each such item was indeed plant- or site-specific. Therefore, the staff finds that Section 3.6 of the AP1000 GTS and Section B 3.6 of the AP1000 Bases are acceptable.

16.4.3.7 Plant Systems

Section 3.7 of the PTS and Bases include requirements for various systems in the secondary side of the SGs (i.e., the main steam safety valves (MSSVs), the main steam isolation valves (MSIVs), the main feedwater isolation valves (MFIVs), etc.), the spent fuel pool water level and makeup systems, and the main control room habitability system. Changes to AP1000 GTS Section 3.7 are described as follows:

- In TS Section 3.7.1, "Main Steam Safety Valves (MSSVs)," the applicant proposed a slight increase in the relief capacity and the resulting relief setpoint for all but the first-to-open MSSV based on a minor change to the valve inlet piping to conform to ASME Code requirements. Also, the applicant replaced the bracketed values for the restriction on maximum allowable thermal power with inoperable MSSVs in Table 3.7.1-1 with new final values. The staff finds the final data in Table 3.7.1-1 acceptable since they were derived using methodology referenced in the Westinghouse STS, Revision 3.

- In addition, the applicant proposed to change the tolerance for the as-found relief setting for MSSVs in Table 3.7.1-2 from 1 percent to 3 percent. In RAI-SRP16-CTSB-11, the staff asked the applicant to provide justification for the change in Table 3.7.1-2. In its response dated December 17, 2008, the applicant proposed to change this tolerance back to the original value of 1 percent. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS Section 3.7.6, "Main Control Room Habitability System," the applicant proposed to use the preliminary bracketed value of 23,443 kPa (3,400 psig) for the required minimum pressure specified in SR 3.7.6.2, as a final value based on latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. The staff found the proposed final value acceptable since it is consistent with relevant information described in AP1000 DCD Section 6.4.2.

- In addition, the staff noted that the AP1000 GTS did not incorporate the NRC-approved TSTF-448, "Control Room Habitability," which was issued to address safety issues identified in GL 2003-001. In RAI-SRP16-CTSB-32, the staff asked the applicant to address these issues. In its response dated November 11, 2008, the applicant stated that it had added a new DCD Section 6.4.5.4, "Main Control Room Envelope Habitability," under Revision 16 to address GL 2003-001. This DCD section describes the periodic testing of the main control room envelope habitability during main control room emergency habitability system operation (pressurization mode) to measure the air inleakage in accordance with the American Society for Testing and Materials (ASTM) E741, "Testing of Palo Verde Units 1-3 Control Room Envelopes." The

applicant concluded that this periodic testing commitment in DCD Section 6.4.5.4, combined with the existing LCO 3.7.6 requirements, adequately addresses the GL 2003-01 issues and provides requirements equivalent to those approved in TSTF-448. The applicant proposed no further changes to the AP1000 DCD or the AP1000 GTS. The staff disagreed with this conclusion. In the May 4, 2009, response letter to RAI-SRP6.4-SPCV-01, the applicant proposed to revise TS 3.7.6 and its associated bases to fully incorporate all TSTF-448 requirements. The staff finds this latest response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. Also in response to RAI-SRP6.4-SPCV-06, the applicant proposed additional design changes to the control room habitability system. As a result, further changes are proposed to TS 3.7.6 and a new administrative control TS 5.5.13 is added to the AP1000 GTS for testing of the new passive filtration unit. Additional open items to TS 3.7.6 and TS 5.5.13 were identified and documented as part of the staff's evaluation of these design changes in SER Section 6.4. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS Section 3.7.9, "Fuel Storage Pool Makeup Water Source," the applicant provided additional information to the precautionary Note 1 in LCO 3.7.9 for clarification. The staff finds the added text acceptable since it is consistent with relevant information described in AP1000 DCD Section 9.1.3.

- At the end of TS Section 3.7, the applicant proposed to add TS 3.7.11, "Fuel Storage Pool Boron Concentration," and TS 3.7.12, "Spent Fuel Pool Storage," to reflect the final design of the spent fuel storage racks. The staff finds the added TS requirements and associated information in the TS bases acceptable since they were formulated in accordance with guidance provided in the Westinghouse STS 3.7.16 and 3.7.17, respectively, and are consistent with relevant information in the AP1000 DCD, Section 9.1. Section 9.1 of this report presents a separate evaluation of the final design of the spent fuel storage racks.

The applicant adhered to the plant systems information as provided in the Westinghouse STS, with differences to reflect AP1000 unique design features. With respect to AP1000 unique design features, the GTS are sufficient to assure operation of these features within the bounds of the safety analysis. In addition, AP1000 GTS Section 3.7 and its Bases do not include "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.7 of the AP1000 GTS and Section B 3.7 of the AP1000 Bases are acceptable.

16.4.3.8 Electrical Power Systems

Section 3.8 of the AP1000 GTS and Bases include requirements for the plant electrical systems that provide redundant, diverse and dependable power sources for all plant operating conditions. In the event of a total loss of offsite power, onsite diesel generators and batteries are provided to supply electrical power equipment necessary for the safe shutdown of the plant. Changes to AP1000 GTS Section 3.8 are as follows:

- Section 3.8, "Electrical Power Systems," of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 16, and TR-134, Revisions 0 through 5, the applicant made minor editorial changes and updated technical information. The staff finds these editorial changes acceptable.

- In the AP1000 DCD, Revision 17, the applicant replaced all preliminary information in the brackets with the final information. The applicant documented the basis for these changes in TR-74, Revision 1. The staff finds these changes acceptable due to the justifications found in the reference report.

- The applicant initially proposed retaining brackets [ ] around all preliminary AP1000 DCD values associated with the battery float current. COL applicants referencing the AP1000 DCD would replace preliminary information, provided in brackets [ ], with final plant specific values. In the AP1000 DCD, Revision 17, the applicant replaced all preliminary information in the brackets with the final information. The staff finds this acceptable since it is consistent with the guidance provided in the Institute of Electrical and Electronic Engineers (IEEE) references.

- The applicant inadvertently omitted the "7 days" completion time in TS Section 3.8.1 B.3 and has added it into AP1000 DCD, Revision 17, and Revision 4 of TR-134. The staff finds this editorial change acceptable.

The applicant adhered to the electrical power systems information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.8 and its Bases do not include "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.8 of the AP1000 GTS and Section B 3.8 of the AP1000 Bases are acceptable.

16.4.3.9 Refueling Operations

Section 3.9 of the AP1000 GTS and Bases includes requirement for boron concentration, unborated water sources, nuclear instrumentation, containment penetrations, and water inventory in the refueling pool during Mode 6. Changes to AP1000 GTS Section 3.9 include the following:

- In TS Section 3.9.5, "Containment Penetrations," the applicant proposed to use the preliminary bracketed information as a final value for the required number of bolts (four) to keep the equipment hatch in place to meet the containment closure requirements during movement of irradiated fuel assemblies within the containment, based on the latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. In RAI-SRP16.1.1-SEB1-01, the staff asked the applicant to provide additional details on the bolt design to ensure the safe installation of the equipment hatch with only four bolts. In its response dated August 15, 2008, the applicant stated that design specification document APP-MV50-Z0-002 would provide final design information for the equipment hatch installation. The discussion and closure of RAI-SRP16.1.1-SEB1-01 is found above in Section 16.4.3.6.

- Also, in TS Sections 3.9.5, "Containment Penetrations"; and 3.9.6, "Containment Air Filtration System (VFS)," the applicant proposed to use the preliminary bracketed value of -0.0311 kPa (-0.125 inches water gauge) relative to outside atmospheric pressure for VFS subsystem testing in SR 3.9.5.3 and SR 3.9.6.3. The applicant proposed using the preliminary value as a final value based on the latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. In RAI-SRP16-CTSB-59, the staff asked the applicant to explain the basis for the selected value. In its response dated August 15, 2008, the applicant stated the following:

This pressure was chosen based on ASHRAE Applications, which recommends at least 0.0124 kPa to 0.0149 kPa (0.05 to 0.06 inches of water) across boundaries when exfiltration or infiltration is minimized. Conservatively, Westinghouse chose a higher pressure difference of 0.0311 kPa (0.125 inches of water).

The staff finds the stated reason acceptable since the selected value is more conservative than the value used in normal industry practices. Therefore, the staff considers RAI-SRP16-CTSB-59 closed.

- In TS Section 3.9.7, "Decay Time," the applicant proposed to change the minimum decay time of 100 hours to 48 hours to make it consistent with the analysis of the fuel handling accident as described in AP1000 DCD Section 15.7.4. The staff finds this change acceptable for the stated reason.

The applicant adhered to the refueling operations information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.9 and its Bases do not include "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.9 of the AP1000 GTS and Section B 3.9 of the AP1000 Bases are acceptable.

16.4.4 Design Features

Section 4.0 of the AP1000 GTS includes other design features not covered in TS Section 3-series, such as the site location, the site maps, and other information related to core design and fuel storage design. Changes to AP1000 GTS Section 4.0 are as follows:

- In TS Section 4.3, "Fuel Storage," the applicant proposed various changes to the description of the fuel storage area to reflect the final design for new and spent fuel storage racks and an increase of the maximum capacity of the spent fuel storage racks from 616 to 889 fuel assemblies. Evaluation of the final design modification is provided separately in Section 9.1 of this report. Furthermore, in RAI-SRP16-CTSB-38 and RAI-SRP16-CTSB-39, the applicant was asked to address inconsistencies between the information provided in TS Section 4.3 and DCD Section 9.1. In its response dated December 2, 2008, the applicant proposed revisions to TS Section 4.3 and DCD Section 9.1 to revolve these inconsistencies. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the design features information as provided in the Westinghouse STS, with differences to reflect AP1000 unique design features. With respect to AP1000 unique design features, the GTS are sufficient to assure operation of these features within the bounds of the safety analysis. In addition, AP1000 GTS Section 4.0 includes "bracketed information" and "Reviewer's Notes." The staff reviewed each piece of "bracketed information" to understand its intent and to determine whether each was site-specific and appropriately deferred to applicants for construction permits or COLs that reference the AP1000 GTS. The staff concluded that each such item was indeed plant- or site-specific. Therefore, the staff finds that Section 4.0 of the AP1000 GTS is acceptable.

16.4.5 Administrative Controls

Section 5.0 of the AP1000 GTS includes provisions, which address various administrative controls related to plant key personnel responsibilities, plant procedures, special programs and reports, etc., to ensure the plant is safely operated. Changes to AP1000 GTS Section 5.0 are described as follows:

- In TS Section 5.4, "Procedures," the applicant proposed to adopt GL 1982-33, "Supplement 1 to NUREG-0737—Emergency Response Capabilities," dated December 17, 1982, as guidance to be used in the development of the plant emergency operating procedures. This is consistent with the STS and acceptable to the staff.

- In TS 5.5, "Programs and Manuals," and in TS 5.6, "Reporting Requirements," the applicant proposed changes to TS 5.5.4, "Steam Generator Program," and to TS 5.6.8, "Steam Generator Tube Inspection Report," to reflect the implementation of the NRC-approved TSTF-449, Revision 4. The staff finds these changes acceptable since implementing TSTF-449 is one acceptable option for addressing safety issues identified in GL 2006-001. However, since TSTF-449 was prepared to address issues involving SG replacements at current operating plants, in RAI-SRP16-CTSB-76, the staff asked the applicant to make one minor adjustment to its proposed changes in TS 5.5.4 to also accommodate SG initial installations at new nuclear power plants regarding the 100-percent tube inspection during the first refueling outage. In its response dated December 2, 2008, the applicant agreed to make the suggested adjustment in a future DCD revision. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 5.5.8, the applicant proposed to use the preliminary bracketed numerical values as final values for acceptance criteria used in various tests on the containment air locks. The staff finds these final selected values acceptable since they are consistent with recommendations provided in the Westinghouse STS.

- In connection to TS 3.7.6 regarding implementation of TSTF-448, Revision 3, to address safety issues identified in GL 2003-01, in RAI-SRP16-CTSB-32, the staff asked the applicant to include the description of the Control Room Habitability Program into AP1000 GTS Section 5.5. Discussions and closure of RAI-SRP16-CTSB-32 are addressed in Section 16.4.3.7 above.

- Also, in TS 5.5.11, "Battery Monitoring and Maintenance Program," the applicant proposed to adopt the preliminary bracketed texts that are applicable to "vented lead-acid" batteries, as the final texts based on latest system design specifications. The staff finds this acceptable since it is consistent with recommendations provided in the Westinghouse STS.

- As stated in Section 16.4.3.3 above, in response to RAI-SRP16.3-CTSB-SCP-1, the applicant stated that all values specified for trip setpoints and allowable values in Tables 3.3.1-1 and 3.3.2-1 will be determined via a SCP specified in Section 5.5.21, "Administrative Controls." This is in accordance with DC/COL-ISG-8, "Technical Specification Information that Combined License Applicants Must Provide in Combined License Application," in determining instrumentation trip setpoints and allowable values in TS. After selection of specific instrumentation, the trip setpoints can be calculated

using the setpoint methodology specified in the SCP; in WCAP-16361, Revision 0. The staff finds the applicant's response to RAI-SRP16.3-CTSB-SCP-1 acceptable. The staff received the applicant's proposed SCP in an RAI response dated February 19, 2010. Initially, the staff found the proposed SCP unacceptable, and communicated their concerns to the applicant. On May 6, 2010, a revised program was submitted to the staff for review. The staff reviewed the revised SCP and found it to be consistent with the recommendations provided in DC/COL-ISG-8. Based on the comprehensive nature of the SCP, the staff believes a COL applicant can calculate all values necessary to complete information found in Section 3.3 of the TS. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the administrative controls information as provided in the Westinghouse STS, except as noted above. In addition, AP1000 GTS Section 5.0 includes "bracketed information" and "Reviewer's Notes." The staff reviewed each piece of "bracketed information" to understand its intent and to determine whether each was site-specific and appropriately deferred to applicants for construction permits or COLs that reference the AP1000 GTS. The staff concluded that each such item was indeed plant- or site-specific. Therefore, the staff finds that Section 5.0 of the AP1000 GTS is acceptable.

16.5 Conclusion

The staff concludes that the changes to the AP1000 GTS and Bases include design-specific parameters and additional TS requirements considered appropriate by the staff. In addition, the staff has compared the additional TS requirements to the relevant NRC regulations, acceptance criteria defined in NUREG-0800, Section 16.0, and other guidance and concludes that the application is in compliance with NRC regulations.

For the reasons set forth above, the staff finds the changes to AP1000 DCD Chapter 16, GTS and Bases, are acceptable and satisfy the requirements of 10 CFR 50.36, 10 CFR 50.36a, and 10 CFR 52.47(a)(11).

17. QUALITY ASSURANCE

17.3 Quality Assurance During the Design Phase

17.3.1 Introduction

Revision 17 of the AP1000 design control document (DCD) proposes to implement the Westinghouse Quality Management System (QMS), Revision 5, for the AP1000 project. The Nuclear Regulatory Commission (NRC) approved Revision 5 to the Westinghouse QMS in a letter dated September 13, 2002. Revision 0 of technical report (TR)-109, APP-GW-GLR-109, "DCD Revision to Incorporate ASME NQA-1-1994 for AP1000," presents the technical justification for implementing the QMS for the AP1000 project. TR-109 justifies standard changes to the AP1000 DCD Sections 3.8, 5.2, 5.4, 17.3, and Appendix 1A. TR-109 also incorporates the American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA) Standard NQA-1–1994, "Quality Assurance Requirements for Nuclear Facility Applications," into the DCD for AP1000.

17.3.2 Evaluation

NRC regulatory guide (RG) 1.28, Revision 3, "Quality Assurance Program Requirements (Design and Construction)," issued August 1985, endorsed NQA-1-1983 as an acceptable method for complying with the provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," with regard to the requisite quality assurance (QA) program for the design and construction phases of nuclear power plants.

In Revision 15 of the AP1000 DCD, the applicant committed to the guidance of ASME NQA-1b-1991. As stated in TR-109, the proposed change would revise the version of NQA-1 referenced and committed to in the AP1000 DCD from NQA-1b-1991 to NQA-1-1994. The NRC staff compared the two versions of NQA-1 and found that the 1994 Edition differed from the 1991 Edition primarily in format. NQA-1-1994 consolidates NQA-1 and NQA-2 into a single multipart document and restructures its format to facilitate use of various parts of the standard. The basic requirements of the 1991 Edition were substantially unchanged. Additionally, the applicant evaluated the changes from NQA-1b-1991 to NQA-1-1994. As documented in TR-109, these changes include: (1) reordering of definitions; (2) addition of functions for which personnel need to be qualified; (3) addition of "siting" to the list of activities affecting quality and the list of quality functions to be audited; (4) addition of a paragraph for inspection requirements; and (5) removal of the allowance related to obsolete drawings. The applicant determined that these changes did not represent a reduction in commitment and did not impact the AP1000 design.

In a letter dated June 6, 2008, the applicant responded to the NRC's request for additional information (RAI)-SRP17.3-CQVP-01 regarding missing changes to Revision 16 of the AP1000 DCD associated with TR-109. Specifically, the NRC staff noted that DCD Sections 3.8.3.6, 5.2.3.4.1, and 5.2.4.6 were not modified as described in TR-109, nor did Revision 4 of TR-134, APP-GW-GLR-134, "AP1000 DCD Impacts to Support COLA Standardization," capture the changes. In its response, the applicant stated that Revision 5 of TR-134 would capture the supportive changes described in TR-109. The NRC staff confirmed

that TR-134, Revision 5, captures the proposed changes. In addition, the NRC staff confirmed that the changes described in TR-109 and Revision 5 of TR-134 are included in Revision 17 of the AP1000 DCD.

As stated above, the applicant currently implements QMS, Revision 5, which commits to NQA-1-1994. In October 2008, the NRC conducted a QA implementation inspection to verify that design activities conducted for the AP1000 project comply with Revision 5 to the QMS and with the requirements of Appendix B to 10 CFR Part 50 and 10 CFR Part 21, "Reporting of defects and noncompliance." The inspectors verified that Revision 5 of the QMS is adequately implemented and identified three nonconformances to the requirements of Appendix B to 10 CFR Part 50. The applicant has responded to the nonconformances and addressed the actions to correct and prevent recurrence of these nonconformances in a letter dated February 3, 2009. The NRC determined that the actions to correct and prevent recurrence of these nonconformances were adequately addressed by the applicant, as described in a letter dated February 20, 2009, describing the NRC's review of the corrective actions.

The NRC staff identified Open Item OI-SRP 17.3-CQVP-01 for additional inspections of the applicant's implementation of the QMS as it relates to the AP1000 project, if necessary. The NRC has determined that additional inspections are not required to support the licensing decision of the AP1000 DCD Revision 17. Open Item OI-SRP 17.3-CQVP-01 is, therefore, closed.

As required by 10 CFR 52.63, "Finality of standard design certifications," the applicant provided justification for TR-109 in its letter dated May 26, 2007. The applicant stated that DCD Revision 16 includes changes to the certified design information that have resulted from inputs provided by the combined license (COL) applicants that will reference the AP1000 DCD. The applicant stated that the consensus group of AP1000 COL applicants has confirmed all of these changes, and the changes will result in an increased standardization of the certification information that all AP1000 COL applicants will adopt. Based on the NRC staff review of QMS, Revision 5, the NRC staff concludes that the adoption of NQA-1-1994 will result in increased standardization of the design certification (DC) information.

17.3.3 Conclusion

Based on the NRC staff's previous approval and subsequent inspection of the implementation of Revision 5 of the QMS, the NRC staff concludes that the applicant's adoption of NQA-1-1994 is acceptable for Revision 17 of the AP1000 DCD. Based on its review, the NRC staff finds that the QMS, as described in the AP1000 DCD Revision 17, meets the criteria of 10 CFR 52.63(a)(1)(vii) and Appendix B to 10 CFR Part 50 and is, therefore, acceptable.

17.4 Reliability Assurance Program During the Design Phase

17.4.1 Introduction

In the amendment to the AP1000 DCD, the applicant proposed changes to the description of the AP1000 reliability assurance program (RAP). The program has two parts. The first, accomplished during the design phase, is called the design reliability assurance program (D-RAP). The second, formerly called the operational reliability assurance program (O-RAP), is no longer required. References to the O-RAP were removed and operational phase reliability assurance activities (OPRAAs) that accomplish the objectives of the RAP after initial fuel load are now described.

The DCD amendment also clarified the association between the maintenance rule template and Nuclear Energy Institute (NEI) document NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52." The DCA added certain structures, systems, and components (SSCs) to the D-RAP and updated the basis for the SSCs inclusion in the D-RAP as a result of updates to the risk analysis and decisions of the applicant's expert panel.

Finally, the DCD amendment proposed a change to the inspection, test, analysis and acceptance criteria (ITAAC). The staff documented its evaluation of this change in Section 17.6, "Tier 1 Information."

In addition to reviewing the amended DCD, the staff reviewed AP1000 combined license standard technical reports TR-117, APP-GW-GLR-117, Revision 1, "Incorporation of the Maintenance Rule," and TR-132, APP-GW-GLN-132, "Changes to D-RAP Component List." The staff also reviewed applicable sections of TR-134. These sections identify changes to Chapter 17 of the AP1000 DCD. Related changes also appear in Chapter 14 and Chapter 16 of DCD Tier 2, as well as in Section 3.7 of DCD Tier 1. This information is generic to the design and is expected to apply to all COL applications that reference the AP1000 DC.

17.4.2 Evaluation

The RAP has two stages. The first is referred to as the D-RAP. The second stage applies to reliability assurance activities for an operating plant; the objective during this stage is to ensure that the reliability of SSCs within the scope of the RAP is maintained during plant operations. Programs that are required by other regulations accomplish these reliability assurance activities. A separate, duplicative program would not be beneficial. Accordingly, Item E of the staff requirements memorandum (SRM) dated June 28, 1995, associated with SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)," dated May 22, 1995, states that no separate O-RAP is required for licensees under 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants."

Instead, the applicants are to incorporate the activities of the RAP after the design phase into existing plant programs. The following are examples of programs that include OPRAAs:

- maintenance rule
- QA
- inservice testing
- inservice inspection
- technical specifications surveillance requirements
- AP1000 investment protection short-term availability controls
- site maintenance

The OPRAA assignment (of activities formerly identified as within the O-RAP) is consistent with the guidance given in SECY-95-132 and with Section 17.4 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." For this reason, the staff finds the OPRAA assignments to be acceptable. This change also meets the requirement of 10 CFR 52.63(a)(1)(vii) in that it "[c]ontributes to increased standardization of the certification information."

The applicant has deleted from the DCD Section 17.4.7.2, "D-RAP, Phase II"; Section 17.4.7.2.1, "Information Available to Combined License Applicant"; and Section 17.4.7.3, "D-RAP, Phase III." The applicant justified these deletions by the incorporation of the generic final safety analysis report (FSAR) template in NEI 07-02A. The staff has endorsed NEI 07-02A and determined that it provides an acceptable method for complying with the requirement in 10 CFR 52.79(a)(15), "Contents of applications; technical information in final safety analysis report." Specifically, FSARs must describe the program for monitoring the effectiveness of maintenance to meet the requirements of 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (also known as the Maintenance Rule), as well as how the program is to be implemented. This will satisfy the acceptance criteria of NUREG-0800 Section 17.6, "Maintenance Rule," and is, therefore, acceptable to the staff. This change also meets the requirement of 10 CFR 52.63(a)(1)(vii) in that it "[c]ontributes to increased standardization of the certification information."

The applicant has deleted four sections of the AP1000 DCD that addressed COL Information Items 17.5.3, 17.5.5, 17.5.6, and 17.5.7.

COL Information Item 17.5.3 states that "The COL applicant or holder will establish [probabilistic risk assessment] PRA importance measures, the expert panel process, and other deterministic methods to determine the site-specific list of SSCs under the scope of the RAP." NEI 07-02A, Section 17.X.1.1, "Maintenance rule scoping per 10 CFR 50.65(b)," addresses this information item. Section 17.X.1.1 provides guidance on PRA insights for the SSCs, the establishment of the expert panel, the process for updating and maintaining the Maintenance Rule scope and SSC classifications, and the use of other deterministic methods for scoping SSCs.

The AP1000 DCD, COL Information Item 17.5.5 states the following:

> The following activities are represented in Figure 17.4-1 as "Plant Maintenance Program." The Combined License applicant is responsible for performing the tasks necessary to maintain the reliability of risk-significant SSCs. Reference 8 [NUREG/CR-5695, "A Process for Risk-Focused Maintenance,"] contains examples of cost-effective maintenance enhancements, such as condition monitoring and shifting time-directed maintenance to condition-directed maintenance.

NEI 07-02A, Section 17.X.3, "Maintenance Rule Program Relationship with Reliability Assurance Activities," addresses this information. Section 17.X.3 provides guidance on the implementation of operational programs for reliability assurance. These programs include QA, Maintenance Rule, maintenance, surveillance testing, inservice inspection, and inservice testing.

COL Information Item 17.5.6 states, "The Maintenance Rule (10 CFR 50.65) is relevant to the Combined License applicant's maintenance activities in that it prescribes SSC performance-related goals during plant operation."

NEI 07-02A addresses performance-related goals throughout the document. Specifically, Section 17.X.1.1 provides guidance on how SSCs in the Maintenance Rule program are evaluated against performance criteria to determine if goals must be established. In Section 17.X.1.2, the guidance states that goals are established for SSCs classified as

(a)(1) status. Section 17.X.1.4 specifies a periodic evaluation of the performance, condition monitoring, goals, and preventive maintenance activities for SSCs in the Maintenance Rule program.

The AP1000 DCD, COL information item 17.5.7 states the following:

> In addition to performing the specific tasks necessary to maintain SSC reliability at its required level, the O-RAP activities include:
>
> - Reliability data base—Historical data available on equipment performance. The compilation and reduction of this data provides the plant with source of component reliability information.
>
> - Surveillance and testing—In addition to maintaining the performance of the components necessary for plant operation, surveillance and testing provides a high degree of reliability for the safety-related SSCs.
>
> - Maintenance plan—This plan describes the nature and frequency of maintenance activities to be performed on plant equipment. The plan includes the selected SSCs identified in the D-RAP.

These bulleted items are covered under NEI 07-02A, Section 17.X.3. The first bulleted item is also captured in Section 17.X.4, "Maintenance Rule Program Relationship with Industry Operating Experience Activities."

The staff considers NEI 07-02A to be an acceptable way to address these COL information items. These information items are considered to have been addressed for COL holders that implement programs consistent with this guidance. This change also meets the requirement of 10 CFR 52.63(a)(1)(vii) in that it "[c]ontributes to increased standardization of the certification information."

Most components included in the D-RAP (listed in Table 17.4-1, "Risk-significant SSCs within the Scope of D-RAP" and in Tier 1, Table 3.7-1, "Risk-Significant Components") are listed on the basis of a computed measure of risk importance. Specifically, a basic event with a risk achievement worth (RAW) greater than 2.0 or a risk reduction worth (RRW) greater than 0.005 is considered important to risk, as are the associated SSCs. SSCs may be assigned to the D-RAP for other reasons (for example, an appropriately constituted expert panel judges that they are important). The D-RAP list identifies only the expert panel as the basis for the inclusion of certain components, which implies that these components do not meet the quantitative risk importance criteria. The staff requested additional information to justify this conclusion in four cases:

(1) Reactor coolant pump switchgear circuit breakers:

ECS ES 31	ECS ES 41	ECS ES 51	ECS ES 61
ECS ES 32	ECS ES 42	ECS ES 52	ECS ES 62

In a letter dated September 5, 2008, the applicant provided additional information to justify the classification of the reactor coolant pump (RCP) without reference to the

RAW. The failures of the RCP switchgear circuit breakers are captured in the following events.

RC1CB051GO	PUMP A FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB052GO	PUMP A FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB053GO	PUMP A FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB054GO	PUMP A FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB061GO	PUMP B FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB062GO	PUMP B FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB063GO	PUMP B FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB064GO	PUMP B FAILS TO TRIP—BREAKER FAILS TO OPEN

In addition, the applicant stated the following:

> Fault tree logic requires all four pumps to shutdown; however, there are two breakers for each pump with only one of two breakers necessary to shutdown one pump. Consequently, CCF of the RCP switchgear circuit breakers is not modeled and, therefore, the RCP circuit breakers are in the D-RAP (Table 17.4-1) only due to the recommendation of the expert panel (EP).

The DCD for the certified design reported that following an event such as a loss-of-coolant accident (LOCA), steam generator tube rupture, or a stuck-open valve in the main steam line, RCPs must trip, or operator action will be required to achieve safe shutdown. Failure to trip RCPs can prevent operation of the core makeup tanks. Consequently, the staff expects a common-cause failure (CCF) to trip all RCPs to be significant, even if it does not lead to core damage in large LOCAs and some medium LOCAs.

This was confirmed in APP-GW-GL-022, "AP1000 Probabilistic Risk Assessment," Revision 8 (the PRA report), which reported the CCF of RCP circuit breakers to open, represented by event RPX-CB-GO (see Table 29-2, "Common-Cause Failure Calculations," page 29-22). Item 36 in Table 50-23 of the same report, "Risk Importances Sorted by Risk Increase," shows RPX-CB-GO to be a member of more than 400 cutsets with a RAW greater than 50. It appears in reactor coolant loop, RCN, and RCT fault trees. The NRC staff identified this as Open Item OI-SRP17.4-SPLA-01.

In a letter dated May 13, 2009, the applicant proposed a revision to Table 17.4-1, indicating that these breakers are identified as risk-significant because of high RAW for CCF. The staff finds this to be acceptable and considers the open item to be resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

(2) 125-volt direct current 24-hour batteries, inverters, and chargers

The batteries provide power for the protection and safety monitoring system and safety-related valves. In a letter dated September 5, 2008, the applicant reported that the risk-significance criteria are not met by the basic events for these components without considering common cause. The risk-significance criteria for including the inverters, batteries, and battery chargers based on the PRA modeling of common cause

for these components support inclusion in the D-RAP. The rationale for inclusion is changed to RAW/CCF for 125-volt dc 24-hour batteries, inverters, and chargers.

A subsequent design modification changed the voltage of the 1E direct current (dc) system to 250 volts direct current (Vdc). The staff's evaluation of this modification is documented in Section 8.3.2 of this safety evaluation. This had no impact on the way the system is modeled in the PRA and so PRA insights and results were not affected. In a letter dated April 20, 2010, the applicant proposed to revise Table 17.4-1 in Tier 2, reflecting the changes to the design and adding the associated 1E 250 Vdc buses to the explicit scope. This makes the table consistent with the amended design and it is, therefore, acceptable to the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

(3) In-containment refueling water storage tank vents

These vents (PXS-MT-03) provide a pathway to vent steam from the in-containment refueling water storage tank (IRWST) into the containment. In a letter dated September 5, 2008, the applicant reported that the method of modeling documented in APP-GW-GL-022 had been revised in the change discussed in TR-102, APP-GW-GLR-102, "AP1000 Probabilistic Risk Assessment Update Report."

The risk significance of IRWST vents based on the new PRA modeling for large-release frequency (LRF) supports inclusion in the D-RAP. The rationale for inclusion now reflects RAW for the IRWST vents. This is acceptable to the staff.

(4) Motor-operated valves in automatic depressurization system stages 1, 2, and 3

In the AP1000 DC PRA, the applicant reported a RAW value of greater than 40 for the event that represents failure-to-open for 32 combinations of three motor-operated valves (MOVs) in automatic depressurization system (ADS) stages 2 and 3.

In a letter dated September 5, 2008, the applicant reported that the criteria for including MOVs in ADS stages 1, 2, and 3 based on the PRA modeling (risk significance) of basic events before consideration of common cause for these components does not support inclusion of these MOVs in the D-RAP.

However, two common-cause groupings model CCFs of the ADS MOVs. One models the CCF to operate the MOVs of the ADS first, second, and third stages. The criteria for including MOVs in ADS stages 1, 2, and 3 based on LRF risk significance of this CCF supports their inclusion in the D-RAP. The other grouping models the CCFs of 32 combinations of three MOVs in stages 2 and 3 to fail to operate. The criteria for including MOVs in ADS stage 2 and 3 based on CDF and LRF risk significance supports inclusion of stage 2 and stage 3 MOVs in the D-RAP.

The rationale for including these MOVs in D-RAP now reflects RAW/CCF for MOVs in ADS stages 1, 2, and 3. This is acceptable to the staff.

Several additions to the D-RAP have been identified on the basis of risk significance in the internal events Level 1 PRA:

- chemical and volume control (CVS) letdown discharge isolation valve inside reactor containment (CVS-PL-V045)
- CVS control letdown discharge isolation valve outside reactor containment (CVS-PL-V047)
- a new diverse actuation system cabinet outside the main control room
- 6,900-volt alternating current buses ECS-ES-1 and ECS-ES-2, which are power buses fed by the onsite diesel generators and offsite power
- normal residual heat removal system (RNS) stop check valves (RNS-PL-V007A/B and RNS-PL-V015A/B) on the discharge of the RNS pumps, which prevent backflow from the reactor coolant system (RCS)

In addition, two components have been added on the basis of the Level 2 PRA:

- RNS check valve (RNS-PL-V013), which provides a flow path from the RNS pumps to the RCS
- RNS check valve (RNS-PL-V056), which provides a flow path from the cask loading pit to the RNS pump inlet

These additions are consistent with DCD Section 17.4.7.1.5, "SSCs to be Included in D-RAP," and thus they are acceptable to the staff.

17.4.3 Conclusion

The applicant proposes changes to the RAP, described in the AP1000 DCA request. Both the removal of references to the O-RAP and the description of OPRAAs that complement the D-RAP to accomplish RAP objectives are consistent with NUREG-0800 Section 17.4. For this reason, they are acceptable to the staff.

The staff has endorsed adoption of NEI 07-02A, which allows closure of certain COL information items. The staff considers COL Information Items 17.5.3, 17.5.5, 17.5.6, and 17.5.7 to be closed for COL holders that implement programs consistent with this guidance.

The applicant added certain SSCs to the D-RAP and updated the basis for SSC inclusion in the D-RAP to include updates to the risk analysis and decisions of the applicant's expert panel. These changes result from proposed design changes and updated risk analysis. They are consistent with the methodology previously approved and, therefore, acceptable to the staff.

These changes meet the requirement of 10 CFR 52.63(a)(1)(vii) in that each one "[c]ontributes to increased standardization of the certification information."

17.6 Tier 1 Information

17.6.1 Information

Certain SSCs were added to the AP1000 D-RAP. The basis for SSC inclusion in the D-RAP was revised in some cases as a result of updates to the risk analysis and decisions of the applicant's expert panel.

In addition, the applicant proposed a change to the ITAAC for D-RAP.

In addition to reviewing the amended DCD, the staff reviewed TR-132. The staff also reviewed applicable sections of TR-134 and APP-GW-GLE-007, "ITAAC Changes" (Impact Document 07). In these, changes to Section 3.7 of Tier 1 of the AP1000 DCD are identified. Related changes also appear in Chapters 14, 16, and 17 of DCD Tier 2. This information is generic to the design and is expected to apply to all COL applications that reference the AP1000 DC.

17.6.2 Evaluation

Several additions to the D-RAP have been identified on the basis of risk significance in the internal events Level 1 PRA:

- CVS letdown discharge isolation valve inside reactor containment (CVS-PL-V045)
- CVS letdown discharge isolation valve outside reactor containment (CVS-PL-V047)
- Normal RNS stop check valves (RNS-PL-V007A/B and RNS-PL-V015A/B) on the discharge of the RNS pumps, which prevent backflow from the RCS

In addition, two components have been added on the basis of the Level 2 PRA:

- RNS check valve (RNS-PL-V013), which provides a flow path from the RNS pumps to the RCS.
- RNS check valve (RNS-PL-V056), which provides a flow path from the cask loading pit to the RNS pump inlet.

The nominal voltage of the 24-hour batteries was changed from 125 Vdc to 250 Vdc. (The staff's review of this modification is documented in Section 8.3.2 of this safety evaluation.)

Finally, the applicant revised the method by which certain SSCs were identified to clarify the rationale for their inclusion.

Subject to confirmation, the additions and changes will be consistent with DCD Section 17.4.7.1.5, "SSCs to be Included in D-RAP." They will be reflected in DCD Tier 1, Section 3.7, "Design Reliability Assurance Program," Table 3.7-1, "Risk-Significant Components," and Table 2.2.1-1 (untitled). The staff finds the proposed changes to Tier 1 will make Tier 1 and Tier 2 consistent and, therefore, they are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In APP-GW-GLE-007, "ITAAC Changes," (Impact Document 07), the applicant proposed an alternative to the existing D-RAP ITAAC. In Table 3.7-3, "Inspections, Tests, Analyses and Acceptance Criteria," the approved existing acceptance criterion is as follows:

> A report exists and concludes that the estimated reliability of each as-built component identified in Table 3.7-1 is at least equal to the assumed reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs.

To this, the applicant proposed to add:

> For an as-built component with reliability less than the assumed reliability an evaluation shall show that the net effect of as-built component reliabilities does not reduce the overall reliability. Or, an evaluation shall show that there is not a significant adverse effect on the core melt frequency or the large release frequency in the PRA applicable to the plant.

The staff could not conclude that the proposed alternative would satisfy the expectations documented in the SRM dated June 28, 1995, for SECY-95-132.

D-RAP ITAAC should provide assurance that the reliability and availability of risk-significant SSCs are consistent with the certified design (subject to deviations and plant-specifics approved in the COL and reflected in the FSAR). In RAI-SRP17.4-SPLA-04, the staff requested that the applicant propose an alternative D-RAP ITAAC that provides reasonable assurance that the plant is designed and will be constructed in a manner that is consistent with the key assumptions and risk insights for risk-significant SSCs within the scope of D-RAP. The NRC staff identified this as Open Item OI-SRP17.4-SPLA-04.

In a letter dated March 30, 2010, the applicant proposed a revised D-RAP ITAAC. The applicant proposed new language in Table 3.7-3, "Inspections, Tests, Analyses and Acceptance Criteria." The proposed design commitment now reads as follows: "The D-RAP ensures that the design of SSCs within the scope of the reliability assurance program (Table 3.7-1) is consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability)." The ITAAC is to be completed by an analysis: "An analysis will confirm that the design of RAP SSCs identified in Table 3.7-1 has been completed in accordance with applicable D-RAP activities. The following acceptance criteria are provided:

> An analysis report documents that safety-related SSCs identified in Table 3.7-1 have been designed in accordance with a 10 CFR 50 Appendix B quality program.
>
> An analysis report documents that nonsafety-related SSCs identified in Table 3.7-1 have been designed in accordance with a program which satisfies quality assurance requirements for SSCs important to investment protection.

The staff finds that satisfying this commitment will provide adequate assurance that the design products used to procure, manufacture, transport, store, install, inspect, and test risk-significant SSCs are traceable to the licensed design. It is consistent with SRM for SECY-95-132. This component of ITAAC complements the others in providing confidence that the as-built plant is, at the time of initial fuel loading, consistent with the certified design. For this reason, it is

acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

17.6.3 Conclusion

Revisions to DCD Tier 1 Table 3.7-1, "Risk-Significant Components," and Table 2.2.1-1 (untitled) have been proposed. With these changes, the Tier 1 information corresponds to the Tier 2 information in Table 17.4-1, "Risk-Significant SSCs within the Scope of D-RAP," as it has been clarified and updated to reflect the amended design. These additions and changes are consistent with the previously approved methodology and, therefore, they are acceptable to the staff.

The staff reviewed the applicant's proposed change to D-RAP ITAAC. The staff finds that the proposed D-RAP ITAAC will provide assurance that the reliability and availability of risk-significant SSCs are consistent with the certified design (subject to deviations and plant specifics approved in the COL and reflected in the FSAR). It is consistent with the SRM for SECY-95-132 and is, therefore, acceptable to the staff.

These changes meet the requirement of 10 CFR 52.63(a)(1)(vii) in that each one "[c]ontributes to increased standardization of the certification information."

18. HUMAN FACTORS ENGINEERING

Westinghouse has submitted information in support of its design certification (DC) amendment application that Westinghouse considers "proprietary" within the meaning of the definition provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390(b)(5). Westinghouse has requested that this information be withheld from public disclosure and the Nuclear Regulatory Commission (NRC) staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff's evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information appears within "square brackets" as follows:

[ ]

The complete text of this chapter, containing proprietary information can be found at Agencywide Documents Access and Management System (ADAMS) Accession Number ML112091879 and can be accessed by those who have specific authorization to access Westinghouse proprietary information.

18.2 Element 1: Human Factors Engineering Program Management

18.2.5 AP1000 Human Factors Engineering Program Plan (no comparable NUREG-1793 section)

18.2.5.1 Summary of Technical Information

The applicant added APP-OCS-GBH-001, Revision 0, "AP1000 Human Factors Engineering Program Plan," issued January 2008 (hereafter referred to as the AP1000 Human Factors Engineering (HFE) Program Plan, as a reference in Section 18.2.6.1 of the design control document (DCD) to establish completion of combined license (COL) Information Item 2-1. In technical report (TR)-90 (APP-GW-GLR-090, Revision 0, "Strategy for the Closure of the AP1000 Design Control Document, Chapter 18, 'Human Factors Engineering (HFE) COL Information Items,'" issued February 2007), the applicant stated that the AP1000 HFE Program Plan captures the technical content discussed in DCD Section 18.2, and describes implementation methods for incorporating HFE into the AP1000 design process. It is designed to serve as an implementation manual for the engineering staff.

18.2.5.2 Staff Evaluation

The staff determined that this additional reference is consistent with Revision 15 of the AP1000 DCD. It describes the scope of the HFE program in terms of each of the elements identified in NUREG-0711, "Human Factors Engineering Program Review Model," issued February 2004. The reference also describes the processes used to incorporate HFE into the AP1000 design process.

18.2.5.3 Conclusion

The staff concludes that this change does not affect the conclusion in Section 18.2.4 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004).

The AP1000 HFE Program Plan continues to implement NUREG-0711 guidelines.

18.2.6 Evaluation of COL Information Item 18.2-1 (no comparable NUREG-1793 section)

18.2.6.1 Summary of Technical Information

COL Information Item 18.2-1 states the following:

> The COL applicant referencing the AP1000 certified design is responsible for the execution of a U.S. Nuclear Regulatory Commission (NRC)-approved HFE Program.

In DCD Revision 17, the applicant stated:

> [The] AP1000 Human Factors Engineering Program Plan fully captures the information certified in Section 18.2 and provides execution guidance for the NRC-approved HFE Program. The ongoing confirmation that the AP1000 HFE Program Plan is being executed as required is demonstrated by fulfillment of the other COL Information Items in Chapter 18. The final confirmation that the HFE Program Plan has been executed will be demonstrated by completion of the ITAAC [Inspections, Tests, Analyses and Acceptance Criteria] (Tier 1 Material, Table 3.2-1, Items 1 to 13).

18.2.6.2 Evaluation

From a program overview perspective, the applicant used Revision 17 to document changes in the status of a number of COL information items and inspection, test, analyses, and acceptance criteria (ITAAC). TRs provided for staff review include the supporting documentation. When the TRs indicate that partial progress has been made and additional work to address information items is ongoing, the staff evaluated redundancy between the COL information item and the ITAAC. If the staff identified redundancy, then the COL action item was closed. In all cases, the review ensured that final design product completion was appropriately tracked. The staff identified all documents used to conclude that the NUREG-0711 criteria were satisfactorily implemented, and Westinghouse docketed the documents.

With respect to COL Information Item 18.2-1, the staff determined that the information item is closed based on the following:

(1) The AP1000 HFE Program Plan is consistent with AP1000 DCD, Revision 15. It describes the scope of the HFE program in terms of each of the NUREG-0711 elements. It provides additional information on where and how the overall design process should use HFE guidance and, thus, provides reasonable assurance that the applicant will implement and undertake the required HFE activities for the AP1000 design at the most appropriate time in the project schedule. This program element requires no additional product development.

(2) COL information items in other sections and the ITAAC listed in Table 3.2-1 address specific HFE design products that require completion. Retaining this generic information item is redundant with the remaining open information items and ITAAC.

(3) The applicant must implement the HFE verification and validation (V&V) program in accordance with ITAAC Table 3.2-1, Item 1 (DCD Revision 17). This includes five specific tasks that validate and verify HFE program implementation and concludes with an as-built inspection of the human-system interfaces (HSIs) as constructed at the time of plant startup. The combination of these actions provides better verification of field implementation than would be accomplished under this COL information item.

18.2.6.3 Conclusion

The staff concludes that COL Information Item 18.2-1 is redundant to existing ITAAC included in Tier 1, Table 3.2-1. Consequently this COL information item is closed. ITAAC Item 1 (DCD Revision 17) will verify the execution of the NRC-approved HFE program.

18.2.7 Evaluation of COL Information Item 18.2-2 (no comparable NUREG-1793 section)

18.2.7.1 Summary of Technical Information

COL Information Item 18.2-2 states the following:

> Specific information regarding the location of the emergency operations facility and emergency operations facility communications will be provided by the Combined Operating License applicant to address the Combined License information requested in this subsection.

18.2.7.2 Evaluation

The applicant stated in TR-134 (APP-GW-GLR-134, Revision 4, "AP1000 Standard COLA Technical Report," issued March 2008) that TR-136 (APP-GW-GLR-136, Revision 1, "AP1000 Human Factors Program Implementation for the Emergency Operations Facility and Technical Support Center," issued October 2007) partially addresses the information requested by this information item. In TR-136, the applicant described the method used to apply the AP1000 HFE Program Plan to technical support centers (TSCs) and emergency operations facilities (EOFs) used to support AP1000 plants and stated that the COL applicant has overall responsibility for the human factors adequacy of the TSC and EOF. In APP-OCS-GGR-110-P, Revision 1, "AP1000 Technical Support Center and Emergency Operations Facility Workshop," issued February 2008, the applicant described in detail how it developed the information in TR-136.

In TR-136 and subsequently in AP1000 DCD Revision 17, the applicant made changes to this COL information item that deleted HFE design responsibilities that were included in the previously approved COL information item in the DCD, Revision 15. In response to request for additional information (RAI)-SRP18-COLP-21, dated July 31, 2009, the applicant removed EOF and TSC location requirements and added responsibilities for EOF and TSC human factors attributes.

Deletion of location requirements is acceptable because the HFE design is not dependent on location. The location of the EOF and TSC is subject to regulatory guidance. This is addressed in Section 13.3, "Emergency Planning," of this report.

The addition of COL responsibility for defining EOF and TSC human factors attributes is consistent with the intent of the original, approved COL information item and ensures that the HFE design outside the scope of the AP1000 DCD is addressed. In a subsequent revision to the AP1000 DCD, the applicant incorporated the RAI response into the DCD text, which resolves this issue.

From the program description provided in TR-136 and APP-OCS-GGR-110-P, the NRC staff noted a well-structured and disciplined assessment of the HFE requirements applicable to the TSC and EOF. The following examples demonstrate how the applicant used the AP1000 HFE Program Plan and appropriate regulations to identify the HFE design requirements of the TSC/EOF:

- Westinghouse and utility personnel worked together to identify the functional requirements for the TSC/EOF. The diverse experience in this group supported a thorough evaluation.

- Westinghouse extracted specific requirements from the AP1000 DCD; the AP1000 HFE Program Plan; NUREG -0711, Revision 2; NUREG-0696, "Functional Criteria for Emergency Response Facilities," issued February 1981; and NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued March 2002. These documents serve as the basis for identifying the TSC/EOF functional requirements. Identification of functional requirements is one of the basic steps required in the AP1000 HFE Program Plan and NUREG-0711. APP-OCS-GGR-110-P provides complete documentation of how Westinghouse identified applicable functions.

- Westinghouse and utility representatives conducted an operating experience review (OER). Application of lessons learned from operating experience is one of the basic steps required in the AP1000 HFE Program Plan and NUREG-0711.

- Westinghouse completed a task analysis incorporating OER results, observations from emergency plan drills at V.C. Summer and Harris nuclear sites, input from emergency procedures from four different utilities, and review comments from both Westinghouse and utility personnel. In TR-136, the applicant stated that it will capture the requirement for this task analysis in the Operational Sequence Analysis-2 (OSA-2) Implementation Plan. A task analysis is one of the basic steps required in the AP1000 HFE Program Plan, NUREG-0711, and OSA-2 incorporates accepted methods for performing task analyses.

- In accordance with TR-136, Section 2.4.4, Westinghouse has identified applicable HSI design guidelines from the AP1000 HSI design guidelines (APP-OCS-J1-002, Revision 0, "AP1000 Human System Interface Design Guidelines") to promote the human factors design adequacy of the TSC/EOF design. This ensures that standard HSI design requirements will be applied to the appropriate elements of HSI design.

- EOF/TSC HFE design elements outside the AP1000 scope are addressed via a COL information item. This provides reasonable assurance that a complete HFE design will be achieved for these emergency facilities.

Based on the activities outlined above, the applicant's use of a tailored approach in applying the AP1000 HFE program to the TSC and EOF is solidly based on NUREG-0711. The applicant has documented the TSC and EOF task analysis results in APP-OCS-JOA-001, "HFE Analysis to Support TSC and EOF Design." Open Item OI-SRP18-COLP-18 tracked completion of the task analysis and documentation of the results. This open item has been addressed by issuance of the report, which satisfactorily summarizes task analysis results associated with the EOF and TSC and is closed.

18.2.7.3 Conclusion

The applicant has developed a sufficient basis for applying a tailored HFE program to the TSC and EOF and has documented the TSC and EOF task analysis results in the APP-OCS-JOA-001 report. The revised COL Information Item 18.2-2 accurately communicates the COL applicant's responsibility for HFE design of the EOF and TSC. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which incorporates the revised COL information.

18.2.8 Evaluation of Tier 1 Information—Design Commitment 3, ITAAC Table 3.2-1 (DCD Revision 15)

18.2.8.1 Summary of Technical Information

ITAAC Design Commitment 3 (DCD Revision 15) reads as follows:

> Design Commitment: The HSI design is performed for the operation and control system in accordance with the HSI design implementation plan.
>
> Inspection, Tests, and Analyses: An evaluation of the implementation of the HSI design.
>
> Acceptance Criteria: A report exists and concludes that the HSI design for the operation and control system was conducted in conformance with the implementation plan and includes the following documents:
>
> - Operation and Control Centers System Specification Document
> - Functional requirements and design basis documents for the alarm system, plant information system, wall panel information system, controls (soft and dedicated), and the qualified data processing subsystems
> - Design guideline documents (based on accepted HFE guidelines, standards, and principles) for the alarm system, displays, controls, and Anthropometrics
> - Design specifications for the alarm system, plant information system, wall panel information system, controls (soft and dedicated), and the qualified data processing subsystems

- Engineering test report document summarizing outcomes of each man-in-the-loop engineering test iteration performed to support HSI design

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.2.8.2 Evaluation

The staff did not find a one-to-one correlation between the list of completed documents in TR-82 (APP-GW-GLR-082, Revision 0, "Execution and Documentation of the Human System Interface Design Implementation Plan," issued May 2007) and the AP1000 DCD, Tier 1, ITAAC Table 3.2, Design Commitment 3 (DCD Revision 15), acceptance criteria. Design documents were not identified for the following areas:

- functional requirements and design-basis documents for the plant information system
- functional requirements and design-basis documents for controls (soft and dedicated)
- functional requirements and design-basis documents for the qualified data processing subsystems

RAI-SRP18-COLP-05 requested clarification of the discrepancy. The applicant's response, dated May 28, 2008, indicated that terminology changes resulted in the inclusion of the areas listed above in the "Distributed Control and Information System" (APP-OCS-J1-010, "AP1000 Display System Functional Requirements"). The staff found that this document includes the functional requirements and design-basis information for the systems listed above. The staff concluded that this change was limited to renaming and reorganizing information to improve clarity and did not affect the intent of the ITAAC.

Open Item OI-SRP18-COLP-01A, identified that the applicant had not completed all the design specifications listed in the ITAAC. These design specifications were subsequently completed along with specifications for systems not listed.

The staff reviewed the completed documents referenced in TR-82, along with the information provided in the RAI, and concluded that these documents appropriately implement the HSI design implementation plan, as described in AP1000 DCD, Revision 17. This included the documents referenced in the ITAAC. Clarity was consistently good across the procedural hierarchy, and the specificity of design requirements had increased in the transition from the functional design level to design specifications. These procedures provide reasonable assurance that the design process will effectively implement standardized HFE design requirements. Based on these results, OI-SRP18-COLP-01A has been satisfactorily addressed and is closed. The staff evaluates the translation of these design documents to a physical design as part of ITAAC Table 3.2-1 Design Commitment 1b (DCD Revision 17).

18.2.8.3 Conclusion

The staff concludes that the proposed change to delete Design Commitment 3 of the ITAAC is supported by the quality of the design documents that have been produced. The design documents provide the level of detail needed to provide reasonable assurance that the HFE design will be effectively implemented within the control room, remote shutdown station, and local control stations. Design Commitment 3 in ITAAC Table 3.2-1 (DCD Revision 15) is closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.5 Element 4: Task Analysis

18.5.5 Evaluation of Operational Sequence Analysis-2 Implementation Plan and Results Summary

18.5.5.1 Summary of Technical information

In AP1000 DCD, Revision 17, the discussion of OSA-2 deleted the description of a specific theoretical model for evaluating operator workload measures, but still committed to conducting an evaluation of the effect of the HSI design and the task demands on operator workload. In TR-81 (APP-GW-GLR-081, Revision 1, "Closure of COL Information Item 18.5-1, Task Analysis," issued May 2007) and in the RAI-TR81-COLP-01 response dated January 29, 2008, the applicant indicated that APP-OCS-J1R-210, Revision 1 "AP1000 Operational Sequence Analysis (OSA-2) Implementation Plan," would identify the most appropriate task analysis methods to use.

18.5.5.2 Evaluation

The staff reviewed the OSA-2 Implementation Plan, which describes the applicant's methods for analyzing the collected task sequence information needed to satisfy the four issues addressed in the DCD: (1) completeness of available information; (2) time to perform tasks; (3) operator workload analysis; and (4) operational crew staffing. The staff concludes that it is acceptable to remove the prescriptive language from the DCD because the applicant provided a robust implementation plan containing detailed information describing how to conduct an OSA analysis, the tasks that should be part of the analysis, and the expected results from the analysis.

The staff also reviewed APP-OCS-J1R-220, Revision B, "AP1000 Operational Sequence Analysis (OSA-2) Summary Report," which describes the results of conducting the activities described in the implementation plan.

18.5.5.3 Conclusion

Based on its review of the implementation plan and the summary report, the staff has determined that sufficient information exists to address the NUREG-0711 review criteria as described in the COL closure section below.

18.5.6 Evaluation of COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-3)

18.5.6.1 Summary of Technical Information

COL Information Item 18.5-1 states the following:

> Combined License applicants referencing the AP1000 certified design will address the execution and documentation of the task analysis implementation plan presented in Section 18.5.

Appendix F of NUREG-1793 broke this COL information item into two pieces and reworded the information item so that it reads:

> FSER Item 18.5.3-3: The staff reviewed the applicant's task analysis at an implementation plan level of detail; finished products to complete the element were not available for review, but the methodology for conducting a complete task analysis was evaluated. The COL applicant will use this methodology to conduct a complete HFE task analysis after design certification.
>
> FSER item 18.5.3-2: The COL applicant will utilize the information from the AP1000-specific task analysis in the development of its procedures and training programs.

This section addresses execution of the task analysis implementation plan, which Westinghouse completed. TR-81 was submitted to document the task analysis results. The report recommends a revision to Tier 1 of the DCD ITAAC to reflect completion of the AP1000 function-based task analysis and provides a basis for closure of COL Information Item 18.5-1. The applicant also provided the OSA-2 Implementation Plan, which describes the methodology used to conduct the second round of OSA.

18.5.6.2 Evaluation

The staff evaluated the information provided by the applicant in the OSA-2 Implementation Plan and the OSA-2 Summary Report against acceptance criteria from NUREG-0711, Revision 2.

NUREG-0711, Section 5.4(1), states Criterion 1 as the following:

> The scope of the task analysis should include the following:
>
> - selected representative and important tasks from the areas of operations, maintenance, test, inspection, and surveillance
> - a full range of plant operating modes, including startup, normal operations, abnormal and emergency operations, transient conditions, and low power and shutdown conditions
> - Human Actions (HAs) that have been found to affect plant risk by means of PRA importance and sensitivity analyses should also be considered risk-important

- where critical functions are automated, all human tasks, including monitoring of the automated system and execution of backup actions if the system fails

Evaluation of Criterion 1

The staff reviewed OSA-2 Summary Report, which provides the results of the OSA-2 for the AP1000 design in accordance with the OSA-2 Implementation Plan. The OSA-2 Implementation Plan summarizes [ ] components and the corresponding [ ] maintenance, test, inspection, and surveillance (MTIS) tasks used for the task analysis. The inclusion of representative and important tasks from the areas of operations, MTIS during OSA-2 implementation satisfies the requirements in the first bullet of NUREG-0711 Criterion 1.

The OSA-2 Implementation Plan identified [ ] risk-important tasks, including tasks during normal operations, emergency and abnormal operations, and shutdown conditions. The inclusion of these tasks within the scope of the task analysis implementation plan satisfies the requirements of the second bullet of NUREG-0711 Criterion 1.

As described in the AP1000 DCD, the applicant performed OSA-2 for a representative subset of tasks including risk-important human actions, risk-important tasks, and tasks that have human performance concerns. The applicant used human reliability analysis (HRA) to identify [ ] scenarios and associated tasks described in the implementation plan as risk-important tasks. This is an acceptable method for identifying risk-important tasks. The applicant used probabilistic risk assessment (PRA) to identify estimated timeframes for completing the tasks, as well as the beginning and ending steps. For example, the plan discusses tasks associated with [ ] an [ ], and a [ ]. These events are considered to be within the design basis, and risk-important tasks are associated with them. The identification and inclusion of these risk-important tasks within the scope of the OSA-2 Implementation Plan satisfies the third bullet of NUREG-0711 Criterion 1.

The OSA-2 Implementation Plan discusses [ ] tasks identified during OSA-1 as having human performance concerns. These [

]. The selection of tasks that have associated human factors concerns demonstrates that the applicant has chosen to analyze critical automated functions. The implementation plan describes the backup actions to be taken in case of a failure. Analysis of automated system failures and backup actions during OSA-2 satisfies the fourth bullet of NUREG-0711 Criterion 1.

The staff has determined that the scope of the task analysis is consistent with NUREG-0711 Criterion 1.

NUREG-0711, Section 5.4 (2), states Criterion 2 as the following:

> Tasks should be linked using a technique such as operational sequence diagrams. Task analyses should begin on a gross level and involve the development of detailed narrative descriptions of what personnel have to do. The analyses should define the nature of the input, process, and output needed by and of personnel.

Evaluation of Criterion 2

Consistent with the NUREG-0711 criterion, the applicant used the [ ] methodology to conduct two rounds of analysis. The first analysis, OSA-1, identified risk-important tasks, [ ] for safe operation and safe shutdown for the AP1000 design. OSA-2 used the tasks identified from OSA-1 to estimate [ ].

The staff reviewed the OSA-1 report titled "AP1000 Operational Sequence Analysis (OSA) Summary Report," Revision 0, (APP-OCS-J1R-120) and the implementation plans for OSA-1 (APP-OCS-J1R-110) and OSA-2, (APP-OCS-J1R-210). These reports describe the applicant's methods for analyzing the collected sequence information needed to satisfy the four issues identified in the DCD:

(1) Completeness of information: Establish the necessary information for successful task performance. The results of this analysis feed into the interface design process to ensure necessary information is available to the operator performing the task activities.

(2) Time to perform tasks: Establish that the operators will be able to complete tasks within the time available. This information is based on assumptions about the time required to access displays, select and actuate controls, etc. The OSA-2 Summary Report discusses that the generally acceptable range of "good" or appropriate operator workload is between 50 and 80 percent. A workload greater than 80 percent indicates a potential overload, while a workload less than 50 percent indicates a potential underload.

(3) Operator Workload: Establish the impact of task requirements and the HSI design on operator workload.

(4) Operational crew staffing: Establish staffing requirements. The results of the operator workload assessment and the identification of time constraints are used to review the adequacy of staffing assumptions, HSI design, task allocation and work organization.

Since the OSA-1 analysis was more general than the OSA-2 analysis, and the information from OSA-1 was used as input into OSA-2, the applicant's task analysis is consistent with NUREG-0711 Criterion 2 that the analysis should begin on a gross level (OSA-1) and become more detailed as the analysis proceeds.

Because of the overlap between Criteria 2 and 3, the staff presents its evaluation of the applicant's task analysis with regard to development of detailed narrative descriptions under Criterion 3 below and addresses the input, process, and output needed by and from personnel.

Based on its evaluation, the staff concludes that the applicant has satisfactorily met NUREG-0711 Criterion 2.

NUREG-0711, Section 5.4(3), states Criterion 3 as the following:

> The task analysis should be iterative and become progressively more detailed over the design cycle. It should be detailed enough to identify information and control requirements to enable specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment.

Evaluation of Criterion 3

Westinghouse conducted OSA-1 and OSA-2 and described these analyses in APP-OCS-J1R-120, Revision 0, and APP-OCS-J1R-220, Revision B, respectively. The staff evaluated these documents and found that OSA-1 focused on specifying the operational requirements for a set of selected tasks. OSA-1 also identified risk-important tasks and thoroughly described the [ ]. An [ ] was developed to show information such as [ ]. The staff determined that OSA-1 lays the foundation for and describes the tasks that were analyzed during OSA-2. These analyses meet the NUREG-0711 criterion that the task analysis should be iterative and become progressively more detailed over the design cycle, because the task analysis was repeated and OSA-2 is more detailed than OSA-1.

The staff reviewed the task analysis documents to determine whether the task analyses conducted were detailed enough to identify information and control requirements to enable specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment. The staff's evaluation addressed the input, process, and output needed by and from personnel (part of Criterion 2 above). OSA-2 Implementation Plan, Section 2.2.1 indicates that once the sequences for analysis have been identified, information related to the tasks in each sequence must be selected, including the following:

- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]

- [ ]
- [ ]
- [ ]

To illustrate this aspect of the task analysis data collection process, the applicant provided an example task analysis for [ ] in the OSA-2 Summary Report. For one risk-important task, "[ ]" the applicant identified (1) [ ].

Based on its review of the process discussed in the applicant's OSA-2 Implementation Plan and the example provided, the staff concludes that the task analyses conducted were detailed enough to identify information and control requirements to enable specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment.

Based on its evaluation, the staff concludes that the applicant has satisfactorily met NUREG-0711 Criterion 3.

NUREG-0711, Section 5.4(4), states Criterion 4 as the following:

> The task analysis should address issues such as:
>
> - the number of crew members
> - crew member skills
> - allocation of monitoring and control tasks to the (a) formation of a meaningful job, and (b) management of crew member's physical and cognitive workload.

Evaluation of Criterion 4

Section 2.3.3 of the task analysis implementation plan describes operator workload analysis as an evaluation of the effect of the HSI design and the demands on operator workload. The methodology used for assessing [

].

Section 2.3.4 of the task analysis implementation plan discusses the analysis of operational crew staffing. When the OSA indicates [

]. The applicant uses the results from the OSA to [

]. The applicant provided an example of the [

].

The staff has concluded that Westinghouse has conducted a thorough task analysis using both OSA-1 and OSA-2 and has described in detail the results from its analysis. The OSA-2 analysis was conducted in accordance with the implementation plan, which addresses issues such as the number of crew members, crew member skills, and allocation of monitoring and control tasks.

NUREG-0711, Section 5.4 (5), states Criterion 5 as the following:

> The task analysis results should be used to define a minimum inventory of alarms, displays, and controls necessary to perform crew tasks based on both task and instrumentation and control requirements.

Evaluation of Criterion 5

The OSA-2 Implementation Plan, Section 2.3.1, states that the [ ]. The HSI design process uses the [ ] of OSA-2. This approach is consistent with the information described in Revisions 15 through 17 of the DCD and satisfies NUREG-0711 Criterion 5.

NUREG-0711, Section 5.4 (6), states the following as Criterion 6:

> The task analysis should provide input to the design of HSIs, procedures, and personnel training programs.

Evaluation of Criterion 6

The OSA-1 analysis identified inputs to the HSI design including display requirements, display design constraints, performance time constraints, inventory (alarms, controls, parameters), and display organization and navigation constraints. OSA-2 is performed as part of the design development process to understand the estimated operator workload, performance time estimates, staffing issues, and error potential associated with each task. The staff concludes that, as with OSA-1, the results of OSA-2 provide input to the design of HSIs by providing a set of requirements and constraints on operator task performance.

In NUREG-1793, Section 18.5.3, the staff identified COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-2), which states, "The COL applicant will use the information from the AP1000-specific task analysis in the development of its procedures and training programs." In response to RAI-SRP18.5-COLP-01, dated May 28, 2008, the applicant referred to Sections 5.6 and 5.7 of the AP1000 HFE Program Plan (APP-OCS-GBH-001), which describes two documents: APP-OCS-GER-031, "The Incorporation of Human Factors Engineering into the Development of the AP1000 Plant Procedures," and APP-OCS-GER-041, "The Incorporation of Human Factors Engineering into the Development of the AP1000 Plant Training Program." According to the applicant's response to the RAI, the purpose of these documents is to capture the operator training and procedure information identified in the task analyses. These reports ensure that information related to training and procedures is identified, recorded, and communicated to those responsible for the development of the training programs. Open Item OI-SRP18-COLP-17 tracked completion of these documents. The staff has reviewed the completed documents and determined that information from OSA-1 and OSA-2 analyses useful to procedures and training program development has been identified, extracted, and compiled

such that it can be used as direct input by procedure and training developers. Accordingly, the staff finds that Criterion 6 has been satisfactorily addressed and the open item is closed.

NUREG-0711, Section 5.4 (7), states Criterion 7 as the following:

> Considerations should be addressed for plant modifications that are likely to affect HAs previously identified as risk-important, cause existing HAs to become risk-important, or create new actions that are risk-important.

Evaluation of Criterion 7

The applicant is not required to address the impact of plant modifications on risk-important HAs because Revision 17 of the AP1000 DCD applies to new plant construction.

18.5.6.3 Conclusion

In its evaluation of Revision 15 of the AP1000 DCD, the staff reviewed the function-based task analysis and OSA-1 results and concluded that the applicant had developed an acceptable task analysis implementation plan to satisfy the NUREG-0711 criteria for task analyses. The COL applicant was expected to use this methodology to conduct a complete task analysis after DC (Reference COL Action Item 18.5.3-3). To close this action item, Revision 17 of the AP1000 DCD referenced additional task analysis documents, which describe an implementation plan for conducting a second operational sequence analysis (OSA-2) and provide a summary report of the OSA-2 results. The OSA-2 Implementation Plan and OSA-2 Summary Report focus on risk-important human actions, tasks with high human performance concerns, and on MTIS activities. Based on its evaluation of Revision 17 of the DCD and the referenced reports, the staff concludes that the applicant's task analysis conforms to all applicable criteria from NUREG-0711, Section 18.5.

18.5.7 Evaluation of COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-2)

18.5.7.1 Summary of Technical Information

COL Information Item 18.5-1 from the DCD does not correlate well with its counterpart, NUREG-1793 Item 18.5.3-2 which states:

> The COL applicant will utilize the information from the AP1000-specific task analysis in the development of its procedures and training programs.

The DCD Information Item 18.5-1 (see previous section) focuses on documentation of task analysis results and the staff identified information item addresses application of that information.

18.5.7.2 Evaluation

In response to RAI-SRP18.5-COLP-01, the applicant referred to Sections 5.6 and 5.7 of the AP1000 HFE Program Plan, which describes two documents (APP-OCS-GER-031 and APP-OCS-GER-041). These documents capture the operator training and procedure information identified in the task analyses. They provide an acceptable vehicle for communicating this information to those responsible for the development of the procedure and

training programs. This directly addresses the DCD related action to document the task analysis results.

Using task analysis results to support procedure program development is satisfactorily addressed in the writer's guides, which are discussed in Section 18.9.5.2 of this report.

Using task results to support training program development is not directly addressed in the DCD. When the DCD Revision 15 safety evaluation report (SER) was prepared, no action was taken to include an additional COL information item to reflect the conclusions in the SER. Addition of the action to DCD Revision 18 is considered unnecessary. The bases material has been made readily available. COL applicants can use this information as appropriate as they develop systems approach to training (SAT) based training programs in accordance with industry and regulatory guidance.

18.5.7.3 Conclusion

This COL information item is closed because APP-OCS-GER-031 and APP-OCS-GER-041 adequately communicate the task analysis results applicable to procedure and training program development. As mentioned earlier, this work was being tracked by Open Item OI-SRP18-COLP-17, which has been closed.

18.5.8 Evaluation of COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1)

18.5.8.1 Summary of Technical Information

COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1) states the following:

> [A] COL applicant referencing the AP1000 certified design will document the scope and responsibilities of each Main Control Room position, considering the assumptions and results of the task analysis.

The applicant submitted TR-52 (APP-GW-GLR-010, Revision 2, "AP1000 Main Control Room Staff Roles and Responsibilities," issued June 2007) as a basis for closing COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1).

18.5.8.2 Evaluation

TR-52 states that the applicant has fully addressed the COL information item. Revision 18 of the DCD incorporates the applicable changes. As described in Section 4.5 of TR-52, the role of the shift technical advisor (STA) for the AP1000 design, including the role of assessing possible significant plant abnormalities observed during normal operations, is consistent with the typical responsibilities of the STA as listed in NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980.

The staff issued RAI-TR52-COLP-12 asking the applicant to further clarify the duties and responsibilities in some key areas, including the reactor operator (RO) and STA roles in communication and coordination. In its response, dated November 16, 2007, the applicant clarified the roles and responsibilities of the RO and STA, describing the responsibilities for all main control room (MCR) ROs to communicate with the MCR supervisor and local equipment operators (EOs) to ensure coordination of local unit evolutions with plant operations. The RAI response also described the responsibilities of the MCR supervisor to maintain awareness of

directions given to the EOs and to evaluate any abnormal conditions or operating concerns reported by either the ROs or the EOs. The OSA-2 Implementation Plan and the OSA-2 Summary Report also address MCR responsibilities. These responsibilities conform to the requirements in 10 CFR 50.54, "Conditions of licenses."

18.5.8.3 Conclusion

The staff finds that TR-52 adequately describes the MCR staff roles. The applicant's response to RAI-TR52-COLP-12 addresses each section of the RAI not addressed in TR-52; including specifying how each STA responsibility matches with the list of 12 responsibilities in Appendix C to NUREG-0737. These documents in combination provide sufficient information to close COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1).

18.5.9 Evaluation of Tier 1 Information—Design Commitment 2, ITAAC Table 3.2-1 (DCD Revision 15)

18.5.9.1 Summary of Technical Information

ITAAC Design Commitment 2 reads as follows:

> Design Commitment: The applicant performs a task analysis in accordance with the task analysis implementation plan.
>
> Inspection, Tests, and Analyses: An evaluation of the implementation of the task analysis will be performed.
>
> Acceptance Criteria: A report exists and concludes that function-based task analyses were conducted in conformance with the task analysis implementation plan and include the following functions:
>
> - Control reactivity
> - Control reactor coolant system (RCS) boron concentration
> - Control fuel and cladding temperature
> - Control RCS coolant temperature, pressure, and inventory
> - Provide RCS flow
> - Control main steam pressure
> - Control steam generator inventory
> - Control containment pressure and temperature
> - Provide control of main turbine
>
> A report exists and concludes that operational sequence analyses (OSAs) were conducted in conformance with the task analysis implementation plan. OSAs performed include the following:
>
> - Plant heatup and startup from post-refueling to 100 percent power
> - Reactor trip, turbine trip, and safety injection
> - Natural circulation cooldown (startup feedwater with steam generator)
> - Loss of reactor or secondary coolant

- Post-loss-of-coolant accident (LOCA) cooldown and depressurization
- Loss of RCS inventory during shutdown
- Loss of the normal residual heat removal system (RNS) during shutdown
- Manual automatic depressurization system (ADS) actuation
- Manual reactor trip via the protection and monitoring system (PMS), via diverse actuation system (DAS)
- ADS valve testing during mode 1

In DCD Revision 17, the applicant deleted this ITAAC because it had completed the work described.

18.5.9.2 Evaluation

The task analysis consists of a function-based task analysis and two OSA analyses (OSA-1 and OSA-2). As documented in its safety evaluation of the AP1000 DCD Revision 15, the staff reviewed the function-based task analysis and OSA-1 results and concluded that these task analyses are complete. As part of the DCD Revision 17 review, the staff reviewed the OSA-2 Implementation Plan and OSA-2 Summary Report. The reports describe the detailed methodology the applicant used to conduct OSA-2, as well as the results and impact on the four issues described in the OSA-2 Implementation Plan: 1) completeness of available information; 2) time to perform tasks; 3) operator workload analysis; and 4) operational crew staffing. As described, the task analysis was used in establishing the basis for the HFE design.

The staff reviewed the OSA-2 Summary Report (APP-OCS-J1R-220), Revision A, which provides the results of OSA-2 for the AP1000 design in accordance with the implementation plan. The implementation plan summarizes [ ] components and the corresponding [ ] MTIS tasks for analysis. The results summary report also describes [ ] scenarios and [ ] associated tasks that were described in the implementation plan and analyzed during OSA-2 implementation. Open Item OI-SRP18-COLP-02A documented that the task analysis had not been completed for all of the MTIS tasks. This work was subsequently completed and submitted for staff review in OSA-2 Summary Report (APP-OCS-J1R-220), Revision B. Revision B of the results summary report includes the following information:

(1) The report summarizes the analysis of the [ ] risk-important MTIS tasks.

Westinghouse includes a description of the [ ] and the task analysis results for the MTIS in its results summary report. [ ] similar to OSA, which provides a [ ]. This analysis uses "[ ]" logic, which enables the evaluator to determine [ ] for the MTIS tasks. This is consistent with the specification in Criterion 2 (in Section 5.4 of NUREG-0711) that the applicant uses a process like OSA. Appendix C to the summary report presents the results of the MTIS analyses.

Section 2.1 of the summary report discusses the [ ] scenarios developed as a basis for the total of [ ] tasks to be analyzed using the OSA-2 methodology. (The [ ] tasks

equate to [ ] scenarios because [ ].) For each scenario, the description in Appendix A to the summary report includes the [ ]. Appendix B to the summary report discusses the results of the analyses.

The summary report briefly describes the OSA-2 analyses of these [ ] scenarios and [ ] tasks. The analyses identified [ ] risk-important tasks and the following [ ].

The OSA-2 Summary Report contains tables giving detailed [ ], as well as the [ ]. The descriptions include the [ ]. For example, each scenario and its related tasks are labeled as "[ ]. In the case of an [ ], the first basic event is [ ]. The beginning step or cue in this case is the [ ]. The second cue is that at [ ]. Westinghouse continues to describe the next few events including the [ ]. This analysis continues until the [ ]. Westinghouse's OSA-2 for this particular task includes [ ]. In this case, each task associated with the basic event [ ].

The staff has concluded that the task analysis for Revision 17 of the AP1000 DCD is complete and has sufficient depth to support control room inventory identification and workload analysis. Open Item OI-SRP18-COLP-02A is closed and Design Commitment 2 in ITAAC Table 3.2-1 (DCD Revision 15) is complete and closed.

18.5.9.3 Conclusion

The staff has reviewed the OSA-2 Implementation Plan, Revision 1, and the OSA-2 Summary Report, Revision B, and has determined that the applicant has adequately addressed the criteria found in Section 5 of NUREG-0711. In addition, the staff's review has determined that there is sufficient information to close COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1), COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-2) and COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-3). Design Commitment 2 in ITAAC Table 3.2-1 (DCD Revision 15) is complete and closed. The task analysis that was completed under this ITAAC provides reasonable assurance that a complete Control Room Inventory has been identified. The task analysis also demonstrates that the HFE design ensures an acceptable workload for the operators.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.7 Element 6: Human Reliability Analysis

The applicant made no substantive changes to this section. However, Westinghouse submitted TR-59 (APP-GW-GLR-011, Revision 0, "AP1000 Standard Combined License Technical Report, Execution and Documentation of the Human Reliability Analysis/Human Factors Engineering Integration") to close COL Information Item 18.7-1.

18.7.5 Evaluation of COL Information Item 18.7-1

18.7.5.1 Summary of Technical Information

COL Information Item 18.7-1 states the following:

> Combined license applicants referencing the AP1000 certified design will address the execution and documentation of the human reliability analysis/human factors engineering integration implementation plan that is presented in Section 18.7.

Westinghouse submitted TR-59 to close COL Information Item 18.7-1. TR-59 summarizes the applicant's method for conducting the HRA/HFE evaluation for the AP1000 and unites the relevant HRA/HFE evaluation implementation plan with the results documentation.

The staff reviewed and approved Westinghouse Commercial Atomic Power (WCAP)-14651, Revision 2, "Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan," as a supporting document for DCD Revision 15. Sections 2 through 5 of WCAP-14651 describe the major aspects of the plan:

- Section 2 discusses the PRA/HRA identification of critical HAs and risk-important tasks.
- Section 3 describes the task analyses for critical HAs and risk-important tasks.
- Section 4 discusses the reexamination of critical HAs and risk-important tasks.
- Section 5 provides information on the validation of HRA performance assumptions.

The staff used this implementation plan (in addition to NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 18, "Human Factors Engineering," Revision 2, issued March 2007, and NUREG-0711, Revision 2) to review WCAP-16555, Revision 1, "AP1000 Identification of Critical Human Actions and Risk Important Tasks." In addition to TR-59, Westinghouse provided WCAP-16555 to the NRC to close COL Information Item 18.7-1. In WCAP-16555, the applicant provided the results of the evaluation of the AP1000 PRA/HRA that identifies the critical HAs and risk-important tasks for plant operation.

18.7.5.2 Evaluation

The staff determined that WCAP-16555 addresses Section 2 of the WCAP-14651 implementation plan. The applicant addressed Sections 3 through 5 of the implementation plan in Parts 1 and 2 of the OSA.

Section 2 of WCAP-14651 relates to Criterion 1 in Section 7.4 of NUREG-0711, which states the following:

> Risk-important human actions should be identified from the PRA/HRA and used as input to the HFE design effort.
>
> - These actions should be developed from the Level 1 (core damage) PRA and Level 2 (release from containment) PRA including both internal and external events. They should be developed using selected (more than one) importance measures and HRA sensitivity analyses to provide

> reasonable assurance that an important action is not overlooked because of the selection of the measure or the use of a particular assumption in the analysis.

Section 2 of WCAP-14651 discusses the PRA/HFE identification of critical HAs and risk-important tasks. Sections 2.1 and 2.2, respectively, describe the process used to identify critical HAs and risk-important tasks. Section 2.2 is divided into three sections describing the process to identify the risk-important quantitative, qualitative, and qualitative MTIS criteria.

Evaluation Criterion 1—Critical Human Actions: Section 2.1 of WCAP-14651 states that the applicant will determine critical HAs using both deterministic and PRA criteria. In Section 3.1 of WCAP-16555, the applicant presented the results of the analyses, which determined that there were no critical actions for the AP1000. For the deterministic criterion, there were no Type A (as defined in Sections 7.5.2.1 and 7.5.3.1 of the DCD) post-accident instruments and no HAs were required to mitigate any design-basis accident. For the PRA criteria, the analysis showed that no HA, when failed in the PRA, results in a core damage frequency (CDF) of $1x10^{-4}$ core damage events per reactor-year or greater. Further, no HA, when failed in the PRA, results in a large release frequency of $1x10^{-5}$ events per reactor-year. Thus, there are no critical actions for the AP1000 plant. This is in accordance with the design objectives of the AP1000.

Evaluation Criterion 2—Quantitative and Qualitative Risk-Importance Criteria: Section 2.2 of WCAP-14651 states that the applicant will use both quantitative and qualitative criteria to identify the risk-important tasks of the AP1000 design. The quantitative criteria are a risk achievement worth (RAW) of 3.0 and a risk reduction worth (RRW) of 1.1. The RAW is a value that examines the increase in risk that would result if a single HA were to fail. The RRW value examines the decrease in risk that would result if an HA were made perfectly reliable for a given process or parameter. The focused PRA reduced these values to an RAW of 2.0 and an RRW of 1.05.

Section 3.2 of WCAP-16555 and related tables provide the results of the evaluation using the RAW and RRW measures. The applicant performed evaluations for both the CDF and the large early release frequency and considered the internal events, flooding, fire, and shutdown PRAs. The applicant identified about 20 risk-important tasks, summarized in Table 3.2-2. The staff also compared the HAs in the dominant sequences with the top operator actions determined by the risk-importance measures. The dominant sequences and the operator actions were consistent. The staff finds that the applicant's use of quantitative risk-importance criteria meets the objective of the implementation plan.

Section 2.2 of WCAP-14651 includes five qualitative criteria for identifying additional risk-important tasks in conjunction with an expert panel. The applicant used the criteria listed in WCAP-16555, Section 2.2.1, to identify the qualitative risk-important HAs. These criteria are consistent with those in the implementation plan. The applicant also provided the results of this evaluation in Sections 2.2.1 and 3.2.1 of WCAP-16555. The expert panel identified three HAs that were added to the list of risk-important tasks. This approach to identifying the qualitative risk-important HAs is consistent with that given in the implementation plan. The staff finds this to be acceptable.

Evaluation Criterion 3—MTIS Risk-Importance Qualitative Criteria: Section 2.2 of WCAP-14651 provides qualitative criteria for identifying risk-important MTISs.

In Section 3.3 of WCAP-16555, the applicant gives the methodology used to identify the MTIS activities for the risk-important structures, systems, and components (SSCs). A group of engineers representing various disciplines and backgrounds, including HFE, HRA, and PRA, reviewed the results produced by this methodology. The applicant also provided Tables 3.3-1 and 3.3-2, which present the results of the MTIS evaluation. Table 3.3-1 includes the initial list of SSCs considered for MTIS activities, along with any other components that may be risk-important and have interfaces with the control room but may not have been included in the initial list. Lastly, Table 3.3-2 lists the representative MTIS activities that will receive the HFE review. In cases where the same MTIS activity was repeated for different SSCs, one of those MTIS activities from that list was chosen to represent (or selected as a "representative" of) that group.

The staff requested clarification in RAI TR-59-11 about the activities outside of the control room and whether they were included in the set of MTIS tasks identified through the expert panel. The staff noted that the Davis-Besse reactor vessel incident is an example of the need for proper MTIS task identification. The reactor vessel is a risk important SSC, and inspection of the vessel exterior would be an MTIS activity that seems worthy of appropriate planning at the design stage to address human factors issues associated with this activity. Thus, by including activities outside of the control room, accessibility can be assured and procedures and training provided to avoid the kinds of problems that occurred with reactor vessel leakage and corrosion. In its response dated July 27, 2007, the applicant provided information clarifying that operator actions outside of the control room were considered and noted that two of the actions considered were outside of the control room. Further, the passive nature of the plant design limits the use of manual control valves, and the manual control valves that are risk-important have main control room position indication.

The staff finds that the applicant has acceptably implemented the process specified in WCAP-14651 to identify the MTIS risk-important tasks.

Criterion 2 in Section 7.4 of NUREG-0711 states the following:

> Risk-important HAs and their associated tasks and scenarios should be specifically addressed during function allocation analyses, task analyses, HSI design, procedure development, and training. This will help verify that these tasks are well supported by the design and within acceptable human performance capabilities (e.g., within time and workload requirements).

Criterion 2 Evaluation

WCAP-14651, Section 3, describes the process for including the HRA risk-important activities in the task analysis. Westinghouse's OSA documents (for OSA-1 and OSA-2) summarize how the applicant input the HRA risk-important tasks into the task analysis. The OSA-1 Summary Report, Table 3-1, specifically addresses the risk-important tasks. The OSA also detailed task sequences and performance requirements. The applicant gave details of its methodology for task identification with regard to emergency operating procedures (EOPs), system operating procedures, and general operating procedures (GOPs). Section 4.2.4 of the OSA-1 Summary Report presents recommendations for the risk-important actions. Finally, in Section 1 of the OSA-1 Summary Report, Westinghouse stated that the results of the OSA are a set of requirements and constraints on operator task performance and that these are fed into the HSI design. The staff finds that the applicant has acceptably implemented the process described in the implementation plan.

Criterion 3 in Section 7.4 of NUREG-0711 states the following:

> The use of PRA/HRA results by the HFE design team should be specifically addressed; that is, how are risk-important HAs addressed (through HSI design, procedural development, and training) under the HFE program to minimize the likelihood of operator error and provide for error detection and recovery capability.

The applicant submitted the implementation methodology for OSA-2 to address part of Sections 3 and 4 in WCAP-14651. The applicant also provided the OSA-2 Summary Report for review. These documents meet the objectives of Sections 3 and 4 of WCAP-14651, by assigning focus areas for operators, by including MTIS activities in OSA-2, and by using operating procedures during the process. The staff finds that the applicant acceptably implemented the process described in the implementation plan.

Criterion 4 in Section 7.4 of NUREG-0711 states the following:

> HRA assumptions such as decision making and diagnosis strategies for dominant sequences should be validated by walkthrough analyses with personnel with operational experience using a plant-specific control room mockup or simulator. Reviews should be conducted before the final quantification stage of the PRA.

WCAP-14651 describes the process for the validation of the HRA performance assumptions. The applicant is implementing this process as part of its integrated system validation for the AP1000. Section 10.0 of this report details the review of the process used to integrate the HRA risk-important HAs.

18.7.5.3 Conclusion

The staff concludes that TR-59 (APP-GW-GLR-011), WCAP-16555, and the related RAI response (RAI-TR59-11) describe an acceptable approach to implementing WCAP-14651 and to meeting the criteria in Section A.6 of NUREG-0800 and Section 7.4 of NUREG-0711. Based on this material COL Information Item 18.7-1 is closed.

18.7.6 Evaluation of Tier 1 Information—Design Commitment 1, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15)

18.7.6.1 Summary of Technical Information

ITAAC Design Commitment 1 reads as follows:

> Design Commitment: The integration of HRA with HFE design is performed in accordance with the implementation plan.
>
> Inspection, Tests, and Analyses: The applicant will perform an evaluation of the implementation for the integration of HRA with HFE design.

Acceptance Criteria: A report exists and concludes that critical human actions (if any) and risk important tasks were identified and examined by task analysis, and used as input to the HSI design, procedure development, staffing, and training.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.7.6.2 Evaluation

This ITAAC was deleted in Revision 16 of the AP1000 DCD (but the number was kept as a place holder), then subsequently removed entirely from Revision 17. For Revision 17 to the DCD, the applicant has provided the methodology and summary reports that show the risk-important tasks were examined and would have input into the other HFE elements listed in the acceptance criteria. Also, the work products provided by the applicant demonstrate the following:

- There are no "critical human actions" because of the AP1000 passive design.
- "Risk-important actions" as well as "significant" actions are identified and included in the HFE design process in accordance with NUREG-0711 guidance.
- The OSA-1 analysis included all identified actions from the HRA. OSA-2 is a reiterative analysis (see Section 18.5 of this report) that also includes input from the HRA.

18.7.6.3 Conclusion

The staff concludes that Design Commitment 1 in ITAAC Table 3.2-1 (DCD Revision 15) is complete and closed and the COL Information Item 18.7-1 is complete and closed because risk-important HAs have been identified in accordance with the implementation plan and these HAs have been appropriately implemented in the HFE design via the task analysis in OSA-2.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.8 Element 7: Human-System Interface Design

18.8.3 General Human System Interface Design Feature Selection

18.8.3.1 Summary of Technical Information

In DCD Section 18.8.1.8, the applicant deleted reference to the use of computer-based models of cognitive response to control room events as an analytic method supporting workload analysis. The applicant substituted the term "task analysis": the sentence now reads, "Analytic methods include the use of task analysis."

18.8.3.2 Evaluation

NUREG-1793 Section 18.8.1.3 discusses task analysis only from a generic perspective as one of the NUREG-0711 elements. NUREG-1793 Section 18.8 does not include specific methods for evaluating workload. In both cases, the change described above does not affect the evaluation or conclusions from this section of the safety evaluation. Section 18.5, "Task Analysis," provides an evaluation of the impact of the change on task analysis.

18.8.3.3 Conclusion

The staff concludes that this change does not affect the evaluation or results documented in NUREG-1793 Section 18.8.1.3.

18.8.4 Evaluation of COL Information Item 18.8-1

18.8.4.1 Summary of Technical Information

COL Information Item 18.8-1 states the following:

> The COL applicant referencing the AP1000 certified design is responsible for the execution and documentation of the HSI design implementation plan.

The applicant issued TR-82 to address this COL information item. In this document, the applicant stated that the COL item has been fully addressed and no additional work is required by the COL applicant.

18.8.4.2 Evaluation

The applicant has satisfactorily completed documentation of the HSI design implementation plan. The staff reviewed the completed documents referenced in TR-82 and concluded that they appropriately execute the HSI design implementation plan, as described in the AP1000 DCD, Revision 15. The specificity of design requirements clearly increased in the transition from the functional design level to design specifications. The documents were consistently clear across this procedural hierarchy. The scope of and specificity in the design documents provide reasonable assurance that the design process will effectively produce the design document needed to support procurement, construction and inspection activities.

This COL information item is redundant to Design Commitment 3 from ITAAC Table 3.2-1 (DCD Revision 16), which states that the HSI design is performed for the operation and control system in accordance with the HSI design implementation plan. Based on this redundancy, the COL information item is closed.

18.8.4.3 Conclusion

The applicant is completing design documents in accordance with the HSI design implementation plan. While the applicant has not completed execution of the HSI design implementation plan, the COL information item is being closed because it is redundant to an existing ITAAC.

18.8.5 Review of Human Factors Evaluation Style Guide (APP-OCS-J1-002) against NUREG-0711 Criteria

18.8.5.1 Summary of Technical Information

The applicant submitted AP1000 HSI Design Guidelines (APP-OCS-J1-002, Revision 0). This document implements several NUREG-0711 criteria that have not been previously reviewed at the implementation plan level. The evaluation below verifies that the AP1000 HSI Design Guidelines effectively address applicable NUREG-0711 criteria.

18.8.5.2 Evaluation

Criterion 1—Style Guide

NUREG-0711, Section 8.4.5, "HSI Detailed Design and Integration Criteria," Criterion 1 states the following:

> Design-specific HFE design guidance (style guide) should be developed. The design of the HSI features, layout, and environment should incorporate HFE guidelines.

In APP-OCS-J1-002, the applicant provided a detailed set of HFE requirements for all HSIs similar to the level of detail in NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines," issued May 2002. The goal of the document is to ensure that the AP1000 designs comply with applicable HFE design principles.

The staff concludes that this document meets this criterion for design-specific HFE guidance.

Subcriterion—Style Guide Content

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The content of the style guide should be derived from (1) the application of generic HFE guidance to the specific application, and (2) the development of the applicant's own guidelines based upon design-related analyses and experience. The applicant may justify guidelines that are not derived from generic HFE guidelines based on an analysis of recent literature, analysis of current industry practices and operational experience, tradeoff studies and analyses, and the results of design engineering experiments and evaluations. The guidance should reflect the applicant's design decisions that address the specific goals and needs of the HSI design.

In APP-OCS-J1-002, the applicant included a list of technical references used to develop specific HFE guidance for the AP1000 design. The applicant used NUREG-0700 as a major source. The following references also support the AP1000 HFE design guidance:

- [ ]
- [ ]
- [ ]

- [ ]
- [ ]
- [ ]
- [ ]

The staff concludes that these technical references represent a diverse and thorough set of inputs for the AP1000 guidance. The AP1000 design guidance includes design principles and specific design criteria for all of the AP1000 HSIs.

Subcriterion—Scope and Level of Detail

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The topics in the style guide should address the scope of HSIs included in the design and address the form, function, and operation of the HSIs, as well as environmental characteristics relevant to human performance.

In APP-OCS-J1-002, Section 3, the applicant described the scope of the design guidelines. This includes the MCR, remote shutdown station, and TSC. Specific HSI interfaces include the plant information system, alarm system, computerized procedures, safety systems, soft controls, dedicated controls, DAS, and large screen displays. The scope addresses all areas described by the previously reviewed program-level documents. APP-OCS-J1-002, Section 26, includes environment-related criteria.

The staff concludes that the design guideline addresses the HSI scope satisfactorily. The level of detail is consistent with that found in NUREG-0700, an accepted program for HFE design criteria.

Subcriterion—Guideline Specificity

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The individual guidelines should be expressed in concrete, easily observable terms. In general, generic HFE guidelines should not be used in their abstract form. Such generic guidance should be translated into more specific design guidelines that can, as much as possible, provide unambiguous guidance to designers and evaluators. They should be detailed enough to permit their use by design personnel to achieve a consistent and verifiable design that meets the applicant's guideline.

The level of detail provided in individual guidelines is consistent with the specificity in NUREG-0700. In general, the guidelines provide quantifiable direction. For many of the guidelines, and particularly for those cases in which more general direction is given, the basis for the guideline is included. This reference provides direction on guideline implementation. The guidance is divided into required and optional categories, which provides additional support to the designers and evaluators.

The staff concludes that the direction provided in the design guidance document is of sufficient detail that design personnel will be able to achieve a consistent and verifiable design.

Subcriterion—Style Guide Ease of Use

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The style guide should provide procedures for determining where and how HFE guidance is to be used in the overall design process. The style guide should be written so that designers can readily understand it. The style guide should support the interpretation and comprehension of design guidance by supplementing text with graphical examples, figures, and tables.

APP-OCS-J1-002 provides generic direction stating that the design guidance will be used during the design process and to facilitate design verification. Implementation plans for both of these activities refer to the use of the [ ]. The plans cross-reference between [ ] and the applicable sections of APP-OCS-J1-002, which will likely facilitate the use of the [ ], as indicated in the criterion, [ ], and the [ ] is provided to answer questions that might arise as to the applicability of the design guidance.

The staff concludes that the design guidance in APP-OCS-J1-002 is presented in a manner likely to facilitate its use by designers and evaluators. The applicant has provided sufficient cross-referencing in procedures to ensure their appropriate use.

Subcriterion—Usability

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The guidance should be maintained in a form that is readily accessible and usable by designers and that facilitates modification when the contents require updating as the design matures. Each guideline included in the guidance documentation should include a reference to the source upon which it is based.

The applicant maintains APP-OCS-J1-002 on its electronic document tracking system as a controlled document. This ensures document accessibility and facilitates usability by virtue of word search capability. The document itself is [ ]. The applicant has demonstrated the ability to keep the document updated by incorporating more detail on [ ]. Each guideline includes a reference to source material; this should also aid the designer in determining how best to implement the requirements and to facilitate the evaluation of tradeoffs.

18.8.5.3 Conclusion

The staff concludes that APP-OCS-J1-002 provides specific HFE design guidance that satisfactorily implements NUREG-0711 criteria. The document provides sufficient detail to ensure that the process is consistently followed and provides reasonable assurance that design requirements are properly factored into the HSIs.

18.9 Element 8: Procedure Development

The applicant made no substantive changes to this section. However, Westinghouse submitted TR-70 (APP-GW-GLR-040, Revision 1, "Plant Operations, Surveillance, and Maintenance Procedures") to close COL Information Item 18.9-1.

18.9.5 Evaluation of COL Information Item 18.9-1

18.9.5.1 Summary of Technical Information

COL Information Item 18.9-1 was identified in NUREG-1793 and does not have a counterpart in the DCD. This COL action item is divided into two parts. The COL action item states the following:

> With regard to procedure development, the COL applicant will (1) address the procedure development considerations in NUREG-0711, and (2) identify the minimum documentation that the COL applicant will provide to the staff to complete its review.

Westinghouse submitted TR-70 for staff review. This report documents the methodology, criteria, and schedules for procedure development. The document addresses the information needed to close COL Information Item 18.9-1. The applicant made the TR-70 supporting documents available to the staff for the purpose of closing COL Information Item 18.9-1. Two of these documents were the writer's guides for normal operating procedures and two-column operating procedures (APP-GW-GJP-100, Revision G, "AP1000 Normal Operating Procedures (NOPs) Writer's Guideline," and APP-GW-GJP-200, Revision D, "Writer's Guideline for Two Column Procedures," respectively). The writer's guidelines explain the programmatic process that controls the preparation of the normal operating procedures and two column procedures.

The goal of the staff's review was to address each part of the action item. Consequently, the evaluation is described in two parts. Part 1 details how the applicant addressed the procedure development considerations in NUREG-0711. Part 2 describes the documents that were submitted to the staff for review.

18.9.5.2 Part 1—Evaluation

The staff reviewed TR-70 in combination with the writer's guides. The staff verified that the applicant had implemented the guidelines specified in WCAP-14690, "Designer's Input to Procedure Development for the AP600." WCAP-14690 is the staff-approved document that describes the methodology the COL applicant should use to develop procedures. In NUREG-1793, the staff approved the use of this document as a guide for procedures development and an acceptable guideline for creation of an implementation plan for the AP1000. In its review, the staff found that the writer's guides meet the criteria in NUREG-0711, Section 9.4, for the basis, development, and content of the AP1000 two column and normal operating procedures. The staff found that the information in TR-70 is consistent with the guidelines in WCAP-14690. Section 2.0 of WCAP-14690 details the general criteria that an applicant should implement to develop procedures. TR-70 addresses all of the guidance criteria in Section 2.0 of WCAP-14690. Section 4.0 of WCAP-14690 provides guidance on the process that should be used to write the plant-specific EOPs. Sections 3.0 and 5.0 of the WCAP describe the guidance for creation of the implementation plan with regard to computer-based

procedures (CBPs). The following section documents the CBP evaluation as a subpart to addressing Part 1 of COL Information Item 18.9-1.

Human Factors Engineering Aspects of Computer-Based Procedures

The applicant did not address the impact of computerized procedures and accessibility in the original DC application. In the staff's evaluation of the AP1000 DCD, NUREG-1793 states the following:

> Evaluation of the applicant's computerized procedure system was not included in the design certification for the AP1000. WCAP-14690, Revision 1, provides information on the computer-based procedure system which will serve as the interface to the plant procedures.

NUREG-0700, Section 8; Interim Staff Guidance (ISG)-05 ("Task Working Group #5: Highly-Integrated Control Room—Human Factors Issues"); and NUREG-0711, Section 9.4, Criteria 7 and 9, are used to evaluate the methodology used to design the CBP system and the interaction between the operator and that system. ISG-05 is used as complementary review guidance for Criterion 9.

Criterion 7 states the following:

> An analysis should be conducted to determine the impact of providing CBPs and to specify where such an approach would improve procedure utilization and reduce operating crew errors related to procedure use. The justifiable use of CBPs over paper procedures should be documented. An analysis of alternatives in the event of loss of CBPs should be performed and documented.

In TR-70 or in the supporting referenced documentation, the applicant addressed the impact and utilization of CBPs not addressed in the original DC application. In Section 2.7 of TR-70, Revision 1, the applicant stated that comments from operations personnel involved in the human factors testing of the AP1000 control room design, and specifically the computerized procedure system, have been generally favorable. The applicant also documented the results of the analysis of the impact of providing CBPs in the referenced report WCAP-14645-NP, Revision 3, "Human Factors Engineering Operating Experience Review Report for the AP1000 Nuclear Power Plant." The staff reviewed WCAP-14645-NP, Revision 3. The applicant identified multiple human performance issues with the CBPs and then noted the solution, or proposed solution, for each issue.

The staff issued RAI-SRP18-COLP-14 to the applicant requesting the analysis of alternatives to CBPs, in the event that a loss of CBPs occurs. In the RAI-SRP18-COLP-14 response dated August 4, 2008, the applicant stated that it would conduct this analysis as part of the second OSA, described in Section 2.1 of APP-OCS-J1R-210.

Subsequent to this RAI, the staff reviewed APP-OCS-J1R-220, Revision B, OSA-2 Summary Report. The OSA-2 Summary Report identifies the [ ]. This task, [ ], has [ ] that are described in Scenario 16. Also, in this section of the summary report, Westinghouse described how the [ ]. Appendix B, Section B.22, to the report gives details of the [ ] steps described in Scenario 16.

Open Item OI-SRP-COLP-19 was established to track an RAI clarifying how a loss of CBPs is managed. In the RAI-SRP18-COLP-19 response, dated September 1, 2009, the applicant provided the staff with this clarification:

- [ ].
- [ ].
- [ ].

Based on this information the open item was closed.

The staff conducted an audit of the CBP interface at the Westinghouse Energy Center in Monroeville, Pennsylvania in September 2009. During the audit the staff reviewed the AP1000 Computerized Procedure System (CPS) design process, including supporting documentation, as well as the characteristics and functions of the current system as implemented in the AP1000 engineering test simulator. The CPS characteristics and functions included [ ].

Based on the audit the staff concluded that the Westinghouse AP1000 CPS system was designed in accordance with the NRC certified HSI Design Implementation Plan and that all supporting documentation was acceptable and consistent with the NRC design review guidance including the guidance specific to CBP systems. The design as currently implemented is consistent with Westinghouse's design procedures and documentation.

Criterion 9 states the following:

> The physical means by which operators access and use procedures, especially during operational events, should be evaluated as part of the HFE design process. This criterion generally applies to both hard-copy and computer-based procedures, although the nature of the issues differs somewhat depending on the implementation.

The staff used ISG-05 as the complementing review guidance for NUREG-0711, Section 9.4, Criterion 9. ISG-05 provides review criteria for how the user will interface with the CBP system. The applicant provided the documentation to satisfy the ISG-05 criteria in APP-OCS-J1-020, Revision A, "Computerized Procedures System Functional Requirements." APP-OCS-J1-020 documents how the operator physically interfaces with the computer procedure system. The technical information in APP-OCS-J1-020 is consistent in addressing the criteria in ISG-05. CBPs are designed to be the primary procedure interface and access is gained via the video display units. Audit observations confirmed that the CBP system is easily accessed from visual display unit (VDU) menus. Navigation to a specific procedure is via a procedure menu. Navigation between procedures is typically driven by embedded links but the operator can also return to the main menu to select the desired procedure. Navigation was found to be simple and straightforward. Use of the hardcopy procedures, which are available in the control room as a backup to the CBPs, followed conventional practices. The staff submitted RAI-SRP18-COLP-11 requesting clarification of the CBP automation and whether the AP1000 computer procedure system would be computer-paced or user-paced. In its

RAI-SRP18-COLP-11 response, dated August 4, 2008, the applicant stated that this issue would not be of any consequence because the computer-paced function would be removed. The staff found this response acceptable.

18.9.5.3 Part 1—Conclusion

The staff determined that TR-70 and the writer's guides for normal and two column procedures together constitute an acceptable implementation plan for procedure development. This is because: (1) the documents address the criteria in the staff-approved WCAP-14690, which explains the process the procedure writer should take to develop an implementation plan; and (2) the documents also address the applicable criterion in the procedures development chapter in NUREG-0711.

The staff concludes that Westinghouse has designed a system that ensures the usability and usefulness of CBPs. Specifically, loss of the CBP HSI is appropriately addressed in procedures and training. Support provided for the transition to paper based procedures provides reasonable assurance that such a failure would not significantly impact the operator's ability to implement the appropriate accident response procedures. Further, the staff concluded that Westinghouse's approach for implementing a new technology into the control room and operating practices was acceptably conservative and should provide for a smooth transition to computerized operation of important procedures, such as EOPs. This approach will minimize any safety concerns associated with the loss of the CPS.

Based on the preceding information, the staff concludes that COL Information Item 18.9-1, Part 1 is complete and closed.

18.9.5.4 Part 2—Evaluation

To address the second part of COL Information Item 18.9-1, in addition to submitting TR-70, the applicant stated in Revision 17 of the AP1000 DCD that the COL applicant will be responsible for addressing the operational and programmatic issues and training to complete the AP1000 COL licensing process. Westinghouse would be responsible for managing the development, review, and approval of the AP1000 normal operating, abnormal operating, emergency operating, refueling and outage planning, alarm response, administrative, and MTIS procedures, as well as the procedures that address the operation of post-72-hour equipment.

18.9.5.5 Part 2—Conclusion

In DCD Tier 2, Revision 17, responsibility for completing this COL action was assumed by Westinghouse. As described above, sufficient documentation has been submitted to satisfy the criteria in NUREG-0711, Section 9.4. COL applicants have continuing responsibilities related to training and procedures but these are evaluated as part of operating program inspections. This Westinghouse response satisfies Part 2 of COL Information Item 18.9-1 (NUREG-1793 Item 18.9.3-1).

The staff concludes that the applicant's procedure development program provides reasonable assurance that procedures will support and guide human interaction with plant systems, as well as control plant-related events and activities. Human engineering principles and criteria are applied, along with all of the other design requirements, to develop procedures that are technically accurate, comprehensive, explicit, easy to use, validated, and in conformance with 10 CFR 50.34(f)(2)(ii). In addition, this closes OI-SRP18-COLP-19.

The staff concludes that COL Information Item 18.9-1, Part 2 is complete and closed. COL Information Item 13.5-1 covers the remainder of the procedures development.

18.11 Element 10: Human Factors Verification and Validation

Westinghouse submitted the following implementation plans to address COL Information Item 18.11-1 and ITAAC Design Commitment 4, Tier 1, Table 3.2-1 (DCD Revision 15):

- APP-OCS-GEH-120, "AP1000 Human Factors Engineering Design Verification Plan," Revision B
- APP-OCS-GEH-220, "AP1000 Human Factors Engineering Task Support Verification Plan," Revision B
- APP-OCS-GEH-320, "AP1000 Human Factors Engineering Integrated System Validation Plan," Revision D (Integrated System Validation (ISV) Plan)
- APP-OCS-GEH-321, "AP1000 Human Factors Engineering Integrated System Validation Scenario Information," Revision B (ISV Scenario Plan)
- APP-OCS-GEH-420, "Human Factors Engineering Discrepancy Resolution Process," Revision B
- APP-OCS-GEH-520, "AP1000 Human Factors Engineering Design Verification at Plant Startup," Revision B

NUREG-0711 states the following:

> An implementation plan gives the applicant's proposed methodology for meeting the acceptance criteria of the element. An implementation plan review gives the applicant the opportunity to obtain staff review of and concurrence in the applicant's approach before conducting the activities associated with the element. Such a review is desirable from the staff's perspective because it provides the opportunity to resolve methodological issues and provide input early in the analysis or design process when staff concerns can more easily be addressed than when the effort is completed.

The staff will verify the final results of the design analyses to ensure that the design is completed in accordance with the process specified in the implementation plans in accordance with the design acceptance criteria (DAC) approach. This may occur via a DC amendment, the COL application review, or through the ITAAC closure process.

When conducting an implementation plan review, the staff needs to:

- understand how the detailed methodology will be implemented
- determine that the methodology can be reliably conducted by design personnel
- be confident that the methodology will provide results that will be acceptable as evaluated by the relevant NUREG-0711 review criteria

18.11.5 Evaluation of COL Information Item 18.11-1

18.11.5.1 Summary of Technical Information

COL Information Item 18.11-1 states the following:

> Combined License applicants referencing the AP1000 certified design will address the development, execution and documentation of an implementation plan for the verification and validation of the AP1000 Human Factors Engineering Program. The programmatic level description of the AP1000 verification and validation program presented and referenced by Section 18.11 will be used by the Combined License applicant to develop the implementation plan.

18.11.5.2 Evaluation

COL Information Item 18.11-1 includes two distinct activities related to the AP1000 HFE program V&V. The first activity addresses development of an implementation plan. Design Commitment 4, Tier 1, Chapter 3, ITAAC Table 3.2-1, of the AP1000 DCD, Revision 15, also addresses this commitment. The second activity is to execute and document the execution of the implementation plan. Design Commitment 5, Tier 1, Chapter 3, ITAAC Table 3.2-1, of the AP1000 DCD, Revision 15, addresses this commitment.

18.11.5.3 Conclusion

The NRC staff notes that COL Information Item 18.11-1 is similar to existing Design Commitments 4 and 5, ITAAC Table 3.2-1 (DCD Revision 15). The development of the implementation plans has been completed and these implementation plans are evaluated below under Evaluation of Tier 1 Information Design Commitment 4 below. The execution and documentation of the implementation plans will be addressed in Design Commitment 5, ITAAC Table 3.2-1. Thus, COL Information Item 18.11-1 is no longer needed since the work has either been completed by Westinghouse or will be completed under the DCD ITAAC 5.

18.11.6 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 1 of 5, HSI Task Support Verification

18.11.6.1 Summary of Technical Information

ITAAC Design Commitment 4 reads as follows:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (part 1): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the ...HSI task support verification activity.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.11.6.2 Evaluation

NUREG-0711, Section 11.4.2.2, Criterion 1, states the following:

> The criteria for task support verification come from task analyses of HSI requirements for performance of personnel tasks.

Evaluation of Criterion 1

In APP-OCS-GEH-220, the applicant provided a specific verification plan for each of the task analysis inputs as outlined below:

- Section 2.2 is the verification plan for the function based task analysis. APP-OCS-J1A-030, Revision A, "FBTA Summary Report," provides [ ]. The [ ]. If the final design does not implement the recommendations, [ ].

- Section 2.3 is the verification plan for OSA-1. These tasks are derived from [ ]. A database is used to maintain the tasks identified by this analysis. Before final task verification, the plan requires the database to be [ ]. The independent verifier ensures that for each unique operator action, [ ].

- Section 2.4 is the verification plan for OSA-2 [ ]. If a new task is identified, then the OCS product manager ensures that disposition of the task is addressed. For each task identified in OSA-2, a list of [ ]. The independent verifier then confirms that the HSI resource is available, the HSI display information is appropriate, the communication facility is available and located appropriately, and the labeling is correct.

- Section 2.5 is the verification plan for the OSA-2 tasks specific to [ ]. Verification follows the same process as that used for OSA-1.

- Section 2.6 is the verification plan for the OSA-2 tasks specific to [ ]. Verification follows the same process as that used for OSA-1.

The staff concludes that APP-OCS-GEH-220 provides clear, specific direction on how the results of each specific task analysis are verified. Acceptance criteria are stated within the procedure and, when combined with the use of an independent verifier, provide reasonable assurance that the HSI requirements properly incorporate the task analysis results.

NUREG-0711, Section 11.4.2.2, Criterion 2, "General Methodology," states the following:

> The HSIs and their characteristics (as defined in the HSI inventory and characterization) should be compared to the personnel task requirements identified in the task analysis.

Evaluation of Criterion 2

The implementation plan for task support verification, as outlined above, provides clear direction that the final HFE design is to be compared to personnel task requirements. Direction is provided to document and justify or resolve all deviations. The direction is structured so that each task is specifically addressed. This supports a clear communication of source documents and acceptance criteria to be used in the verification.

The staff concludes that APP-OCS-GEH-220 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for the general methodology of task verification.

NUREG-0711, Section 11.4.2.2, Criterion 3, states the following:

> Human engineering discrepancies (HEDs) should be identified when an HSI needed for task performance is not available or when HSI characteristics do not match personnel task requirements.

Evaluation of Criterion 3

In APP-OCS-GEH-220, the applicant stated that any time an HSI resource or an appropriate display is not available; a discrepancy worksheet is filled out. The procedure specifically states the following verification points:

- [ ].
- [ ].
- [ ].
- [ ].
- [ ].

When the V&V evaluation is complete, the OCS product manager assesses each work discrepancy worksheet. Discrepancies that are directly justified as exceptions are not considered HEDs. The applicant documents justified discrepancies as part of future report APP-OCS-GER-120, "AP1000 HFE Task Support Verification Report," along with a list of HEDs identified by discrepancy reports. APP-OCS-GEH-420 provides an implementation plan for resolving the discrepancy worksheets that are not justified by the product manager.

The staff concludes that APP-OCS-GEH-220 provides sufficient details of the implementation plan to satisfactorily demonstrate implementation of this NUREG criterion for identifying task requirement deficiencies during task verification.

NUREG-0711, Section 11.4.2.2, Criterion 4, states the following:

> An HED should be identified for HSIs that are available in the HSI, but are not needed for any task....

Evaluation of Criterion 4

In APP-OCS-GEH-220, Sections 2.3.2 (OSA-1) and 2.4.2 (OSA-2), the applicant stated that the independent verifier will check each display for information and/or controls that are not

associated with task requirements. Deviations must be documented on a discrepancy worksheet.

The staff concludes that APP-OCS-GEH-220 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for identifying unnecessary HSI components during task verification.

18.11.6.3 Conclusion

The staff concludes that APP-OCS-GEH-220 provides an implementation plan that satisfactorily implements the guidance in NUREG-0711 relative to task support verification. The level of detail provided and the use of an independent verifier provides reasonable assurance that the HSI requirements properly incorporate the results from all task analyses performed. This element of ITAAC Design Commitment 4 (DCD Revision 15), as described above, is complete and closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.11.7 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 2 of 5, HFE Design Verification

18.11.7.1 Summary of Technical Information

ITAAC Design Commitment 4 reads as follows:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (Part 2): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the …HFE Design Verification activity.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.11.7.2 Evaluation

NUREG-0711, Section 11.4.2.3, Criterion 1, states the following:

> The HFE guidelines serve as review criteria. Selection of specific guidelines depends on the characteristics of the HSI components included in the scope of review and whether the applicant has developed a design-specific guideline document. NUREG-0700 may be used for HFE design verification.

Evaluation of Criterion 1

In APP-OCS-GEH-120, the applicant stated that HSI resources and operation and control centers are verified against APP-OCS-J1-002. APP-OCS-J1-002 satisfactorily implements NUREG-0711, Section 8.4.5(1), as described in Section 18.8. It includes guidance from NUREG-0700, Revision 2, and Commission Electrotechnique Internationale/International Electrotechnical Commission (CEI/IEC) 964, "Design for Control Rooms of Nuclear Power Plants," which program-level documents specifically cite. The report also includes results from operating experience review, function-based task analysis, and other industry guidance.

The staff concludes that APP-OCS-GEH-120 provides sufficient direction to ensure that the HFE guidelines serve as review criteria and have an appropriate level of detail. The report is also consistent with the program description.

NUREG-0711, Section 11.4.2.3, Criterion 2, states the following:

> The applicant should compare the characteristics of the HSI components with the HFE guidelines to determine whether the HSI is acceptable or discrepant (i.e., an HED).
>
> The applicant should evaluate discrepancies as potential indicators of additional issues.

Evaluation of Criterion 2

In APP-OCS-GEH-120 the applicant provided a complete list of [ ] (Section 1.2.2). The general process description in Section 2.1 specifies that each [ ] APP-OCS-J1-002. Appendices B and C provide [ ]. APP-OCS-J1-002 provides pass/fail criteria. A discrepancy worksheet documents all discrepancies. Disposition of discrepancies can be handled immediately by the OCS product manager or submitted to the AP1000 HFE engineering discrepancy resolution process described in APP-OCS-GEH-420. A future report, APP-OCS-GER-120, will describe all discrepancies and their justification or resolution.

The staff concludes that the implementation plan provides a disciplined process for verifying that the HSI design effectively implements design acceptance criteria. Discrepancies are documented and subjected to a corrective action process that evaluates the potential for additional issues. The staff concludes that APP-OCS-GEH-120 provides sufficient detail to satisfactorily demonstrate implementation of this NUREG criterion for design verification methodology.

NUREG-0711, Section 11.4.2.3, Criterion 3, states the following:

> The applicant should document HEDs in terms of the HSI component involved and explain how the characteristics depart from a particular guideline.

The evaluation of this criterion is contained in the evaluation of Criterion 2, directly above.

18.11.7.3 Conclusion

The staff concludes that APP-OCS-GEH-120 provides an implementation plan that satisfactorily implements the NUREG-0711 criteria associated with design verification. The document provides reasonable assurance that the HSI designs reflect the design requirements. This element of ITAAC Design Commitment 4 (DCD Revision 15), as described above, is complete and closed.

18.11.8 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 3 of 5, Integrated System Validation

18.11.8.1 Summary of Technical Information

At the time of the Westinghouse AP1000 DC, based on Revision 15 of the DCD, HFE V&V was reviewed and found acceptable at a programmatic level. The Westinghouse V&V program was described in WCAP-15860, "Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan," Revision 2, dated October 2003. Pursuant to Section 18.11.1 of the AP1000 DCD, a COL applicant referencing the AP1000 is committed to developing an implementation plan for V&V consistent with the NRC approved programmatic description in WCAP-15860. ITAAC Design Commitment 4 (Tier 1 Section 3.2, Human Factors Engineering, Table 3.2-1) states:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (part 3): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the ...Integrated System Validation activity.

To fulfill this commitment, Westinghouse has submitted WCAP-16769-P, "AP1000 Human Factors Engineering Verification and Validation," and two implementation plans:

- APP-OCS-GEH-320 (ISV Plan)
- APP-OCS-GEH-321 (ISV Scenario Plan)

Open Item OI-SRP18-COLP-03A was created by the staff to track the review of these documents.

18.11.8.2 Evaluation

The purpose of this review is to determine whether the applicant's ISV Plan and its companion document, the ISV Scenario Plan, provide an acceptable implementation plan in accordance with NUREG-0711. These documents are evaluated using WCAP-15860 and the NUREG-0711 review criteria for operational condition sampling and ISV.

18.11.8.2.1 Applicable Review Criteria

When the staff has an NRC-certified, programmatic-level description of an HFE activity, the review criteria used to evaluate an implementation plan come from two sources: the certified, programmatic description and NUREG-0711. The programmatic description, WCAP-15860, identifies the general ISV approaches and constraints. The staff's review of the ISV Plan's compliance with WCAP-15860 is discussed in Section 18.11.8.2.2 below.

NUREG-0711 criteria were used to evaluate the detailed methodology (taking into account the approved approach described in the WCAP). The ISV review criteria used were from the following sections of NUREG-0711:

- Section 11.4.1 - Operation Condition Sampling
  - Sampling Dimensions (3 review criteria)
  - Identification of Scenarios (2 review criteria)

- Section 11.4.3 - Integrated System Validation
  - Test Objectives (1 review criteria)
  - Validation Test Beds (9 review criteria)
  - Plant Personnel (4 review criteria)
  - Scenario Definition (3 review criteria)
  - Performance Measurement (5 review criteria)
  - Test Design (9 review criteria)
  - Data Analysis and Interpretation (5 review criteria)
  - Validation Conclusions (2 review criteria)

In this document, the NUREG-0711 criteria are used to assess the completeness of the ISV Plan and its acceptability as an implementation plan. The results of the staff's evaluation of the ISV Plan with respect to the NUREG-0711 criteria are provided in Sections 3 and 4 for operational condition sampling and ISV methodology, respectively.

18.11.8.2.2 Compliance with the WCAP-15860

The staff evaluated whether the ISV Plan was developed in accordance with the commitments made in WCAP-15860 and whether the ISV Plan satisfies the NRC review criteria of NUREG-0711, Section 11. In general, the ISV Plan follows the commitments made in WCAP-15860. Inconsistencies noted in earlier revisions of the ISV Plan were documented in RAI-22 and the specific details have now been acceptably addressed in Revision D of the ISV Plan.

Additionally, Section 1.5 of the ISV Plan now states that the ISV Plan conforms to the commitments, scope, purpose, and issues as stated in WCAP-15860, with the exception of two areas where exceptions have been taken. The staff has reviewed these two exceptions and found them acceptable for the reasons stated below.

Exception 1: WCAP-15860 states that ISV will utilize currently qualified operating crews as the participants. However, as AP1000 is a new plant design, the ISV participants will not be fully qualified and experienced AP1000 operators. The ISV subjects will not have the same task performance proficiency as that of fully qualified AP1000 operators.

Evaluation: The staff finds this exception acceptable because the ISV Plan continues to include operating experience specifications that are sufficient to ensure valid testing. The ISV will be relatively more demanding and thus a more conservative test of the HFE design. Section 18.11.8.2.4.3, "Plant Personnel," of this report provides additional detail.

Exception 2: WCAP-15860 states that ISV will address all of the EOPs. However, the ISV Plan states that the ISV scenarios will include a representative subset of the EOPs. The ISV scenarios will ensure that all functional operator knowledge, skills, and abilities addressed in the EOPs are assessed.

Evaluation: The staff finds this exception acceptable because:

- The applicable NUREG-0711 review criteria do not call for 100 percent coverage of procedures during ISV. Section 18.11.8.2.3, "Compliance with NUREG-0711 - Operational Conditions Sampling (OCS)," of this report provides additional detail.

- Westinghouse verified the following:

> The ISV scenarios will ensure that all functional operator knowledge, skills and abilities addressed in the AP1000 EOPs are examined and validated in ISV. While the ISV scenarios may not explicitly cause the operators to enter each of functional recovery procedures, the demand to perform similar EOP steps will be represented [ ]

Additionally, prior to the ISV, [ ]. It also ensures a thorough ISV process. Thus, this exception is acceptable.

18.11.8.2.3 Compliance with NUREG-0711 - Operational Conditions Sampling (OCS)

NUREG-0711, Section 11.4.1, states, "The sampling methodology will identify a range of operational conditions to guide V&V activities. The review of operational conditions sampling considers the dimensions to be used to identify and select conditions and their integration into scenarios."

The objective of reviewing operational condition sampling is to verify that the applicant has identified a sample of operational conditions that: (1) includes conditions that are representative of the range of events that could be encountered during operation of the plant; (2) reflects the characteristics that are expected to contribute to system performance variation; and (3) considers the safety significance of HSI components. These sample characteristics are best identified through the use of a multidimensional sampling strategy to provide reasonable assurance that variation along important dimensions is included in the V&V evaluations.

The staff reviewed the defined scenarios in the ISV Plan and the ISV Scenario Plan to determine whether the OCS dimensions were addressed. The aspects of the specified OCS, both from WCAP-15860 and from NUREG-0711 have been addressed by the ISV Plan and the ISV Scenario Plan.

18.11.8.2.3.1 Sampling Dimensions

The sampling dimensions addressed in NUREG-0711, Section 11.4.1.2, include plant conditions, personnel tasks, and situational factors known to challenge personnel performance.

(1) The following plant conditions should be included:

- normal operational events including plant startup, plant shutdown or refueling, and significant changes in operating power

- failure events, e.g.,
  - instrument failures [e.g., safety-related system logic and control unit, fault tolerant controller, local "field unit" for multiplexer (MUX) system, MUX controller, and break in MUX line] including instrumentation and control (I&C) failures that exceed the design basis, such as a common mode I&C failure during an accident

  - HSI failures (e.g., loss of processing and/or display capabilities for alarms, displays, controls, and computer-based procedures)

- transients and accidents, e.g.,
  - transients (e.g., turbine trip, loss of off-site power, station blackout, loss of all feedwater, loss of service water, loss of power to selected buses or MCR power supplies, and safety and relief valve transients)

  - accidents (e.g., main steam line break, positive reactivity addition, control rod insertion at power, anticipated transient without scram, and various-sized LOCAs)

  - reactor shutdown and cool down using the remote shutdown system

- reasonable, risk-significant, beyond-design-basis events, which should be determined from the plant-specific PRA

- consideration of the role of the equipment in achieving plant safety functions [as described in the plant safety analysis report (SAR)] and the degree of interconnection with other plant systems. A system that is interconnected with other systems could cause the failure of other systems because the initial failure could propagate over the connections. This consideration is especially important when assessing non-Class 1E electrical systems.

Evaluation of Criterion (1)

WCAP-15860, Section 4.6 includes an extensive and multi-dimensional set of criteria that address this particular criterion. The ISV Plan was developed based on this criterion and includes [ ] separate scenarios, which are detailed in the ISV Scenario Plan. ISV Plan, Section 5.1.1, "Events," lists the various [ ] scenarios. Also, the ISV Scenario Plan has an Appendix A titled, "Scenario Specifications" that provides the [ ] scenarios. The scenarios themselves follow in the rest of Appendix A. Additionally, the ISV Scenario Plan, Appendix E, "Tasks of Special Interest," lists each [ ] and in which scenario(s) it is addressed. Appendix E also provides a cross reference between scenarios and the [ ]. This scenario information was compared with the commitments of WCAP-15860 and with the NUREG-0711 criteria. The ISV Plan and ISV Scenario Plan together

were found to satisfy both the programmatic plan and NUREG-0711 and meet the criterion on plant conditions.

(2) The following types of personnel tasks should be included:

- Risk-significant HAs, systems, and accident sequences - All risk-important HAs should be included in the sample. These include [those] identified in the PRA and those identified as risk-important in the SAR and NRC's SER. Situations where human monitoring of an automatic system is risk-important should be considered. Additional factors should be sampled that contribute highly to risk, as defined by the PRA, including:
  - dominant human actions (selected via sensitivity analyses)
  - dominant accident sequences
  - dominant systems (selected via PRA importance measures such as RAW or RRW)

- OER-identified difficult tasks—The sample should include all personnel tasks identified as problematic during the applicant's review of operating experience.

- Range of procedure guided tasks—These are tasks that are well defined by normal, abnormal, emergency, alarm response, and test procedures. The operator should be able to, as part of rule-based decision-making, understand and execute the specified steps. Regulatory guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, Appendix A, includes several categories of "typical safety-related activities that should be covered by written procedures." The sample should include appropriate procedures in each relevant category:

  - administrative procedures
  - general plant operating procedures
  - procedures for startup, operation, and shutdown of safety-related systems
  - procedures for abnormal, off normal, and alarm conditions
  - procedures for combating emergencies and other significant events
  - procedures for control of radioactivity
  - procedures for control of measuring and test equipment and for surveillance tests, procedures, and calibration
  - procedures for performing maintenance
  - chemistry and radiochemical control procedures

- Range of knowledge-based tasks—these are tasks that are not as well defined by detailed procedures. Knowledge-based decision-making involves greater reasoning about safety and operating goals and the various means of achieving

them. A situation may call for knowledge-based decision-making if the rules do not fully address the problem, or the selection of an appropriate rule is not clear. An example in a pressurized water reactor plant may be the difficulty in diagnosing a steam generator tube rupture (SGTR) with a failure of radiation monitors on the secondary side of the plant because: (1) there is no main indication of the rupture (the presence of radiation in secondary side); and (2) the other effects of the rupture (i.e., slight changes in pressures and levels on the primary and secondary sides) may be attributed to other causes. While the operators may use procedures to treat the symptoms of the event, the determination that the cause is an SGTR may warrant situation assessment based on an understanding of the plant's design and the possible combinations of failures that could result in the observed symptoms. Errors in rule-based decision-making result from selecting the wrong rule or incorrectly applying a rule. Errors in knowledge-based decision-making result from mistakes in higher-level cognitive functions such as judgment, planning, and analysis. The latter are more likely to occur in complex failure events where the symptoms do not resemble the typical case, and thus, are not amenable to pre-established rules.

- Range of human cognitive activities—The sample should include the range of cognitive activities performed by personnel, including:
  - detection and monitoring (e.g., of critical safety-function threats)
  - situation assessment (e.g., interpretation of alarms and displays for diagnosis of faults in plant processes and automated control and safety systems)
  - response planning (e.g., evaluating alternatives for recovery from plant failures)
  - response implementation (e.g., in-the-loop control of plant systems, assuming manual control from automatic control systems, and carrying out complicated control actions)
  - obtaining feedback (e.g., of the success of actions taken)
- Range of human interactions—the sample should reflect the range of interactions among plant personnel, including tasks that are performed independently by individual crew members and tasks that are performed by crew members acting as a team. These interactions among plant personnel should include interactions between:
  - MCR operators (e.g., operations, shift turnover walkdowns)
  - MCR operators and auxiliary operators
  - MCR operators and support centers (e.g., the TSC and the emergency offsite facility)

- MCR operators with plant management, NRC, and other outside organizations

- Tasks that are performed with high frequency.

Evaluation of Criterion (2)

As stated in the evaluation of Criterion (1) above, ISV Plan, Section 5.1.1, and ISV Scenario Plan, Appendix E, list the various evolutions (both high frequency and less common tasks), transients, accidents, and risk-important HAs that are included in the [ ] scenarios. The ISV Scenario Plan, Appendix A, provides the [ ] actual scenarios. This scenario information was compared with the commitments of WCAP-15860 and with the NUREG-0711 criteria. The ISV documents satisfy both documents.

The risk-important HAs and tasks are identified in TR-59/WCAP-16555. WCAP-16555, Section 3.2, identifies [ ] post-accident risk-important HAs in Table 3.2-2. The ISV Plan draft included essentially all of these [ ] risk-important HAs in scenarios. However, the risk-important HA to [ ] was excluded in the ISV scenario Plan. An RAI-SRP18-COLP-53 was written. In the ISV Scenario Plan, this risk-important HA is now included as part of Scenario 18, "Loss of RNS during Mid-Loop Operation." Thus, all risk-important HAs are now addressed in at least one ISV scenario (a few are included in two scenarios). This risk-important HA also has local aspects that cannot be adequately simulated as part of ISV. Thus, the actual verification of acceptability of planned local actions associated with the [ ] will need to be deferred until the plant is built. Therefore, the RAI response proposes adding this to the HFE Design Verification at Plant Startup, APP-OCS-GEH-520. The staff reviewed APP-OCS-GEH-520, Revision B, submitted in a letter dated August 2, 2010, and found that verification of local control action has been added to the document.

The ISV Scenario Plan, Appendix F, previously listed the PRA risk-dominant systems for AP1000. This appendix was deleted from Revision B but is still available in the AP1000 PRA. These systems were verified to all be addressed in the scenarios. [ ].

The following important tasks identified from OSA analyses were included in the ISV, as shown in the ISV Scenario Plan, Appendix E, Table E-1: "Loss of DDS"; [ ]. The following OER important tasks were also identified and included in the ISV Table E-1 and the ISV scenarios: [ ].

The scenarios presented in WCAP-15860 and the ISV Scenario Plan were also found to adequately address a broad range of: procedure–guided tasks, human cognitive activities, and human interactions. Thus, the ISV scenarios were found to adequately address the types of personnel tasks specified in the NUREG-0711 Criterion 2 pending confirmation of RAI-SRP18-COLP-53.

(3) The sample should reflect a range of situational factors that are known to challenge human performance, such as:

- Operationally difficult tasks—The sample should address tasks that have been found to be problematic in the operation of nuclear power plants (e.g., procedure versus situation assessment conflicts). The specific tasks selected should reflect

the operating history of the type of plant being validated (or the plant's predecessor).

- Error-forcing contexts—Situations specifically designed to create human errors should be included to assess the error tolerance of the system and the capability of operators to recover from errors should they occur.

- High-workload conditions—The sample should include situations where human performance variation due to high workload and multitasking situations can be assessed.

- Varying-workload situations—The sample should include situations where human performance variation due to workload transitions can be assessed. These include conditions that exhibit: (1) a sudden increase in the number of signals that must be detected and processed following a period in which signals were infrequent; and (2) a rapid reduction in signal detection and processing demands following a period of sustained high task demand.

- Fatigue and circadian factors—The sample should include situations where human performance variation due to personnel fatigue and circadian factors can be assessed.

- Environmental factors—The sample should include situations where human performance variation due to environmental conditions such as poor lighting, extreme temperatures, high noise, and simulated radiological contamination can be assessed.

Evaluation of Criterion (3)

The ISV Plan and ISV Scenario Plan address operationally difficult tasks as identified via the OSA analyses and through the OER. These are summarized in the ISV Scenario Plan, Appendix F, and are discussed in the review of Criterion (2) above. The scenarios have [ ]. This is described in the ISV Plan, Section 5.1.3, "Complications." These complications also are added to the transient or accident scenarios to [ ].

The applicant took exception to addressing fatigue and circadian factors, relying on APP-OCS-GEH-320, Section 5.1.3, which states that the ISV does not address fatigue and circadian factors since it is considered to be impractical to attempt to mimic the conditions that are typical on the operating site. The staff agrees with this position and notes that 10 CFR Part 26, "Fitness for duty programs," Subpart I addresses managing fatigue.

[ ]. For example, [ ]. Other environmental factors are addressed as part of APP-OCS-GEH-520.

Thus, the ISV plans acceptably address Criterion 3.

18.11.8.2.3.2 Identification of Scenarios

The results of the sampling should be combined to identify a set of scenarios to guide subsequent analyses. A given scenario may combine many of the characteristics identified by the operational event sampling.

Evaluation of Criterion (1)

The [ ] scenarios have been developed for use in ISV. The scenarios are quite varied and they do combine the various characteristics outlined in the operational event sampling. Detailed scenario descriptions are provided in the ISV Scenario Plan. The documents reviewed satisfy this criterion.

The scenarios should not be biased in the direction of over representation of the following:

- scenarios for which only positive outcomes can be expected
- scenarios that for integrated system validation are relatively easy to conduct administratively (scenarios that place high demands, data collection or analysis are avoided)
- scenarios that for integrated system validation are familiar and well structured (e.g., which address familiar systems and failure modes that are highly compatible with plant procedures such as "textbook" design-basis accidents)

Evaluation of Criterion (2)

As noted above under Criterion 1, a robust set of [ ] scenarios has been developed and they are described in the ISV Scenario Plan. These scenarios have many failure events both as the key item of the scenario and as peripheral issues. They are not limited to familiar, typical, or easy-to-conduct scenarios or those with only positive outcomes. The documents reviewed satisfy this criterion.

18.11.8.2.4 Integrated System Validation

The objective of reviewing integrated system validation methodology is to verify that the applicant's methodology will validate the integrated system design (i.e., hardware, software, and personnel elements) using performance-based tests that will determine whether it acceptably supports safe operation of the plant.

18.11.8.2.4.1 Test Objectives

(1) Detailed objectives should be developed to provide evidence that the integrated system adequately supports plant personnel in the safe operation of the plant. The test objectives and scenarios should be developed to address aspects of performance that are affected by the modification [of the] design, including personnel functions and tasks affected by the modification. The objectives should be to:

- Validate the role of plant personnel.

- Validate that the shift staffing, assignment of tasks to crew members, and crew coordination (both within the control room as well as between the control room and local control stations and support centers) is acceptable. This should include validation of the nominal shift levels, minimal shift levels, and shift turnover.

- Validate that for each human function, the design provides adequate alerting, information, control, and feedback capability for human functions to be performed under normal plant evolutions, transients, design-basis accidents, and selected, risk-significant events that are beyond-design basis.

- Validate that specific personnel tasks can be accomplished within time and performance criteria, with a high degree of operating crew situation awareness, and with acceptable workload levels that provide a balance between a minimum level of vigilance and operator burden. Validate that the operator interfaces minimize operator error and provide for error detection and recovery capability when errors occur.

- Validate that the crew can make effective transitions between the HSIs and procedures in the accomplishment of their tasks and that interface management tasks such as display configuration and navigation are not a distraction or undue burden.

- Validate that the integrated system performance is tolerant of failures of individual HSI features.

- Identify aspects of the integrated system that may negatively affect integrated system performance.

Evaluation of Criterion (1)

The objectives of the AP1000 ISV are identified in Section 4.2 of WCAP-15860, which has been approved by NRC as part of the original AP1000 DC. They included:

1. Establish the adequacy of the integrated HSI for achieving HFE program goals
2. Confirm allocation of function and the structure of tasks assigned to personnel
3. Validate the EOPs and associated HSI
4. Confirm the dynamic aspects of the HSI for task accomplishment
5. Evaluate and demonstrate error tolerance to human and system failures
6. Establish the adequacy of staffing and of the HSI to support staff to accomplish their tasks

These objectives have been included in Section 1.2 of the ISV Plan. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.2 Validation Testbeds

(1) Interface Completeness—The testbed should completely represent the integrated system. This should include HSIs and procedures not specifically required in the test scenarios. For example, adjacent controls and displays may affect the ways in which personnel use those that are addressed by a particular validation scenario.

Evaluation of Criterion (1)

(2) ISV Plan, Section 2, indicates that the ISV will be performed at a dedicated, purpose-built facility. The facility will employ a high fidelity, near full-scope simulator to represent the AP1000 systems and the MCR. This simulator will satisfy the general requirements of Sections 3 and 4 of American National Standards Institute/American National Standard (ANSI/ANS)-3.5-1998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination."

NUREG-0711 indicates the use of ANSI/ANS-3.5-1998 is an acceptable testbed. This satisfies Criterion 1.

(2) Interface Physical Fidelity—A high degree of physical fidelity in the HSIs and procedures should be represented, including presentation of alarms, displays, controls, job aids, procedures, communications, interface management tools, layout and spatial relationships.

Evaluation of Criterion (2)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1). In addition, in the ISV Plan, Section 5.1.2, "Procedures," the applicant states that the following types of procedures for AP1000 are incorporated into ISV Scenario Plan, and will be used:

- Optimal Recovery
- Functional Recovery
- Shutdown Procedures
- Normal Operating Procedures (NOPs)
- Abnormal Operating Procedures (AOPs)
- Refueling and Outage Procedures
- Alarm Response Procedures (ARPs)
- Maintenance and Surveillance Guidelines
- [ ]
- [ ]
- [ ]

This approach acceptably meets the staff's review criterion.

(3) Interface Functional Fidelity—A high degree of functional fidelity in the HSIs and procedures should be represented. All HSI functions should be available. High functional fidelity includes HSI component modes of operation, i.e., the changes in functionality that can be invoked on the basis of personnel selection and/or plant states.

Evaluation of Criterion (3)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1).

(4) Environment Fidelity—A high degree of environment fidelity should be represented. The lighting, noise, temperature, and humidity characteristics should reasonably reflect that

expected. Thus, noise contributed by equipment, such as air handling units and computers should be represented in validation tests.

Evaluation of Criterion (4)

Due to the constraints of the building, the simulator does not include the passive cooling fins; instead, there is a conventional office building tiled ceiling. This results in the lighting system being somewhat different, although it is still representative of the final lighting system design. In addition, the heating and ventilation is provided by a conventional office building system and is, therefore, not representative of the final as-built MCR. Also, the acoustic properties cannot be completely replicated, although they will be similar (i.e., painted walls, hard ceiling tiles). The simulator will be as representative as possible of the final MCR design, so that the design can be assessed. The applicant believes the differences will have minimal or no impact on ISV crew performance. [ ]. Also, the applicant noted in its response to RAI-SRP18-COLP-49, dated February 2, 2010, that the environmental conditions will be fully assessed in APP-OCS-GEH-520. This approach provides sufficient environmental fidelity for ISV and acceptably meets Criterion 4.

(5) Data Completeness Fidelity—Information and data provided to personnel should completely represent the plant systems monitored and controlled from that facility.

Evaluation of Criterion (5)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1).

(6) Data Content Fidelity—A high degree of data content fidelity should be represented. The information and controls presented should be based on an underlying model that accurately reflects the reference plant. The model should provide input to the HSI in a manner such that information accurately matches that which will actually be presented.

Evaluation of Criterion (6)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1).

(7) Data Dynamics Fidelity—A high degree of data dynamics fidelity should be represented. The process model should be capable of providing input to the HSI in a manner such that information flow and control responses occur accurately and in a correct response time; e.g., information should be provided to personnel with the same delays as would occur in the plant.

Evaluation of Criterion (7)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1).

(8) For important actions at complex HSIs remote from the MCR, where timely and precise HAs are required, the use of a simulation or mockup should be considered to verify that human performance requirements can be achieved. (For less risk-important HAs or where the HSIs are not complex, human performance may be assessed based on analysis such as task analysis rather than simulation.)

Evaluation of Criterion (8)

The use of local control stations (LCSs) and the Remote Shutdown Workstation (RSW) are included in the ISV scenarios. Scenario 7 for [ ] is included in the ISV Scenario Plan. The RSW will be validated using [ ]. Operators will be able to [ ]. ISV Plan, Section 2.1, "Physical Scope and Fidelity", describes the details of the simulated RSW panel. Other LCSs are also included in the ISV Scenario Plan. The one local action that is risk important relates to [ ]. The actual verification of acceptability of planned local actions [ ] will need to be deferred until the plant is built. The staff reviewed APP-OCS-GEH-520 against the response to RAI-SRP18-COLP-53 R1 received in a letter dated May 21, 2010. The staff confirmed the document conforms to the RAI response.

(9) The testbeds should be verified for conformance to the testbed characteristics identified above before validations are conducted.

Evaluation of Criterion (9)

Section 2.3 of the ISV Plan describes the simulator testing to be performed prior to ISV evaluations. The ISV Plan references the ISV Scenario Plan for detail concerning how the testing will be performed. That information is provided in ISV Scenario Plan, Appendix C, "Simulator Testing." The objective of this simulator testing in preparation for ISV is to demonstrate that the simulator responds in a manner similar to the reference unit while utilizing the operating procedures and that it meets ANSI/ANS 3.5-1998. The testing will be carried out [ ]. This includes an estimated [ ] of testing. In addition, the ISV Plan, Section 3.3 describes pilot testing of each ISV scenario to ensure simulator readiness, and to confirm effective functioning of test protocols and data collection. This will be done by personnel different from the test subjects.

This provides an acceptable and comprehensive approach to testbed verification. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.3 Plant Personnel

(1) Participants in the validation tests should be representative of actual plant personnel who will interact with the HSI, e.g., licensed operators rather than training or engineering personnel.

Evaluation of Criterion (1)

Section 4.9 of the ISV Plan indicates that validation crews will consist of currently qualified operating crews. The applicant takes partial exception to this item from the program plan, as described in the ISV Plan, Section 1.5, Item 1. Since the AP1000 is a new plant design, the ISV participants will not be fully qualified and experienced AP1000 operators. The ISV subjects will not have the same task performance proficiency as that of fully qualified AP1000 operators. This is reasonable for ISV, and it does make the ISV somewhat more demanding. In accordance with Section 4.1, "Subjects," of the ISV Plan, the ISV subjects will be comprised of the following:

1. A group that has completed the [ ].

2. A group that has partially completed the [ ]. These subjects will have undertaken [ ] training program. Subjects will comprise [ ]. They will have completed [ ] of AP1000 systems training and [ ] of procedure/simulator based training.

3. A limited group of [ ].

Thus, The ISV crews are samples taken from the crews of the AP1000 customer utilities, as described above. This approach acceptably meets the staff's review criterion.

(2) To properly account for human variability, a sample of participants should be used. The sample should reflect the characteristics of the population from which the sample is drawn. Those characteristics that are expected to contribute to system performance variation should be specifically identified and the sampling process should provide reasonable assurance that variation along that dimension is included in the validation. Several factors that should be considered in determining representativeness include: license and qualifications, skill/experience, age, and general demographics.

Evaluation of Criterion (2)

Section 4.1.1 of the ISV Plan discussed participant selection. Two utilities will be providing [ ] crews each. A set of criteria will be provided to the utilities to guide selection of crew members. The criteria include successful completion of training, age range, skills and abilities range, qualifications variation, and prior experience variation. In addition, no operators participating in previous AP1000 tests will be used. This approach acceptably meets the staff's review criterion.

(3) In selection of personnel, consideration should be given to the assembly of minimum and normal crew configurations, including shift supervisors, reactor operators, shift technical advisors, etc., that will participate in the tests.

Evaluation of Criterion (3)

In Section 4.1.2, "Crew Size and Number," of the ISV Plan, the applicant states that the typical crew size for ISV will be [ ]. The crew size will also be varied as a complication in specific scenarios. The maximum control room staff is also specified in Section 4.1.2 and consists of [ ] personnel. The maximum staffing level will be addressed by [ ]. The actual staffing level is specified for each scenario in the ISV Scenario Plan, Section A.n.3, "Scenario Participants" (for n = 1 to [ ]). Scenario participants vary from [ ] to [ ]. This reasonably addresses minimum, normal and maximum staffing levels, plus other values in between, and is acceptable.

In its RAI-SRP18-COLP-26 response of July 12, 2010, the applicant committed to delete TR-52 from the DCD and replace it with APP-OCS-GJR-003, Revision 2, "AP1000 Main Control Room Staff Roles and Responsibilities" to document new staffing values. This is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

(4) To prevent bias in the sample, the following participant characteristics and selection practices should be avoided:

- participants who are [ ]
- participants in [ ]
- participants who are selected for some specific characteristic, such as using crews that are identified as good or experienced.

Evaluation of Criterion (4)

As described in ISV Plan, Section 4.1, "Subjects"; Section 4.1.1, "Selection"; and Section 4.1.2, "Crew Size and Number", the applicant will use COL utility personnel as test participants and not design personnel. Section 4.1 states that care will be taken to ensure that the test participants do not obtain any prior knowledge of the scenarios to be used in ISV. The participants will not include subjects that participated in the HFE Tests. Section 4.1.1 states that participating utilities will be requested to assign typical crews for ISV testing based on availability. Crews will not be selected for ISV based on individual characteristics. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.4 Scenario Definition

(1) The operational conditions selected for inclusion in the validation tests should be developed in detail so they can be performed on a simulator. The following information should be defined to provide reasonable assurance that important performance dimensions are addressed and to allow scenarios to be accurately and consistently presented for repeated trials:

- description of the scenario and any pertinent "prior history" necessary for personnel to understand the state of the plant upon scenario start-up
- specific initial conditions (precise definition provided for plant functions, processes, systems, component conditions and performance parameters, e.g., similar to plant shift turnover)
- events (e.g., failures) to occur and their initiating conditions, e.g., time, parameter values, or events
- precise definition of workplace factors, such as environmental conditions
- task support needs (e.g., procedures and technical specifications)
- staffing objectives
- communication requirements with remote personnel (e.g., load dispatcher via telephone)

- the precise specification of what, when and how data are to be collected and stored (including videotaping requirements, questionnaire and rating scale administrations)

- specific criteria for terminating the scenario.

Evaluation of Criterion (1)

The ISV Scenario Plan provides scenario descriptions for each of the [ ] scenarios to be part of ISV. Each scenario in the document includes the following:

- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]

Each scenario provides the above information in acceptable detail. At this time, only three scenarios have complete observer guides. These three provide an acceptable example of how the remaining observer guides will be completed. This approach acceptably meets the staff's review criterion.

(2) Scenarios should have appropriate task fidelity so that realistic task performance will be observed in the tests and so that test results can be generalized to actual operation of the real plant.

Evaluation of Criterion (2)

This criterion is addressed through the use of a simulation facility for ISV that satisfies the general requirements of Sections 3 and 4 of ANSI/ANS-3.5-1988; use of COL plant operating personnel in training for operations; and use of realistic but challenging scenarios. This approach conforms to NUREG-0711 Criterion 2.

(3) When evaluating performance associated with operations remote from the MCR, the effects on crew performance due to potentially harsh environments (i.e., high radiation) should be realistically simulated (i.e., additional time to don protective clothing and access radiologically controlled areas).

Evaluation of Criterion (3)

The ISV Plan notes that the use of LCSs is included in the ISV scenarios and they will use simulated interactions with local operations that extend beyond the MCR. Scripted responses will be provided for the operations support staff to perform specified roles as plant personnel in applicable scenarios (e.g., local operators). This is acceptable.

There is one local control action that is a risk-important HA. This is [ ] and is included in an ISV scenario. The actual verification of acceptability of planned local actions associated with the hatches will need to be deferred until the plant is built. This is an acceptable approach. Therefore, the RAI response proposes adding this to APP-OCS-GEH-520. The staff reviewed APP-OCS-GEH-520, Revision B, submitted in a letter dated August 2, 2010, and found that verification of local control action has been added to the document.

18.11.8.2.4.5 Performance Measurement

The review of performance measurement covers measurement characteristics, performance measure selection, and performance criteria.

18.11.8.2.4.5.1 Measurement Characteristics

(1) Performance Measurement Characteristics—Performance measures should acceptably exhibit the following measurement characteristics to provide reasonable assurance that the measures are of good quality (it should be noted that some of the characteristics identified below may not apply to every performance measure):

- Construct Validity—A measure should accurately represent the aspect of performance to be measured.
- Diagnosticity—A measure should provide information that can be used to identify the cause of acceptable or unacceptable performance.
- Impartiality—A measure should be equally capable of reflecting good as well as bad performance.
- Objectivity—A measure should be based on phenomena that are easily observed.
- Reliability—A measure should be repeatable; i.e., if the same behavior is measured in exactly the same way under identical circumstances, the same measurement result should be obtained.
- Resolution—A measure should reflect the performance at an appropriate level of resolution, i.e., with sufficient detail to permit a meaningful analysis.
- Sensitivity—A measure's range (scale) and the frequency of measurement (how often data are collected) should be appropriate to the aspect of performance being assessed.
- Simplicity—A measure should be simple both from the standpoint of executing the tests and from the standpoint of communicating and comprehending the meaning of the measures.
- Unintrusiveness—A measure should not significantly alter the psychological or physical processes that are being investigated.

Evaluation of Criterion (1)

Section 6 of the ISV Plan describes the performance measures to be used to evaluate integrated system performance. The characteristics of the measures are addressed in Section 6.2. Several of the measures are well-known, commonly used measures with established, acceptable measurement characteristics, such as the [ ]. For others developed by Westinghouse, the basis of the measures is identified. For example, a measure of team performance will be used that is based on a [ ]. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.5.2 Performance Measure Selection

(1) A hierarchal set of performance measures should be used, which includes measures of the performance of the plant and personnel (i.e., personnel tasks, situation awareness, cognitive workload, and anthropometric/physiological factors). Some of these measures could be used as "pass/fail" criteria for validation and the others to better understand personnel performance and to facilitate the analysis of performance errors. The applicant should identify which are in each category.

Evaluation of Criterion (1)

Section 6.1 of the ISV Plan describes the measures to be used. The measures are hierarchal including [ ]. Thus, an acceptable hierarchal set of performance measures will be used to assess integrated system performance.

Section 6.3.1 of the ISV Plan identifies the measures to be used as pass/fail (P/F) criteria. P/F measures are measures reflecting [ ]. This provides a reasonable set of measures to serve as P/F criteria.

Performance measures to be used as diagnostic measures are discussed in ISV Plan, Section 6.3.2. The measures are listed in Table 6.3-2 and include all measures collected during ISV trials with the exception of the P/F measures.

This approach acceptably meets the staff's review criterion.

(2) Plant Performance Measurement—Plant performance measures representing functions, systems, components, and HSI use should be obtained.

Evaluation of Criterion (2)

Section 6 of the ISV Plan discusses ISV performance measurement. P/F plant-level measures involve applicable technical [ ]. Plant-level diagnostic measures are also defined for each scenario so that [ ]. For example, for the [ ]. This approach acceptably meets the staff's review criterion.

(3) Personnel Task Measurement—For each specific scenario, the tasks that personnel are [needed] to perform should be identified and assessed. Two types of personnel tasks should be measured: primary (e.g., start a pump), and secondary (e.g., access the pump status display). Primary tasks are those involved in performing the functional role of the operator to supervise the plant; i.e., monitoring, detection, situation assessment,

response planning, and response implementation. Secondary tasks are those personnel [need to] perform when interfacing with the plant, but which are not directed to the primary task, such as navigation and HSI configuration. This analysis should be used for the identification of potential errors of omission.

- Primary tasks should be assessed at a level of detail appropriate to the task demands. For example, for some simple scenarios, measuring the time to complete a task may be sufficient. For more complicated tasks, especially those that may be described as knowledge-based, it may be appropriate to perform a more fine-grained analysis such as identifying task components: seeking specific data, making decisions, taking actions, and obtaining feedback. Tasks that are important to successful integrated system performance and are knowledge-based should be measured in a more fine-grained approach.

- The measurement of secondary tasks should reflect the demands of the detailed HSI implementation, e.g., time to configure a workstation, navigate between displays, and manipulate displays (e.g., changing display type and setting scale).

- The tasks that are actually performed by personnel during simulated scenarios should be identified and quantified. (Note that the actual tasks may be somewhat different from those that should be performed). Analysis of tasks performed should be used for the identification of errors of commission.

- The measures used to quantify tasks should be chosen to reflect the important aspects of the task with respect to system performance, such as:

  - time
  - accuracy
  - frequency
  - errors (omission and commission)
  - amount achieved or accomplished
  - consumption or quantity used
  - subjective reports of participants
  - behavior categorization by observers

Evaluation of Criterion (3)

Section 6 of the ISV Plan discusses ISV performance measurement. Measurement of operator tasks involves both P/F and diagnostic variables. Successful performance of risk-important HAs is a P/F variable. For example, for the [ ] as tasks to assess using P/F measures. For diagnostic purposes, the performance of key tasks is measured. These actions are listed in the observer guides for each scenario. This approach acceptably meets the staff's review criterion.

(4) Situation Awareness—Personnel situation awareness should be assessed. The approach to situation awareness measurement should reflect the current state-of-the-art.

Evaluation of Criterion (4)

Section 6 of the ISV Plan discusses ISV performance measurement. Situation awareness is measured using the [ ]. SART is a widely used and acceptable measure of situation awareness. This approach acceptably meets the staff's review criterion.

(5) Cognitive Workload—Personnel workload should be assessed. The approach to workload measurement should reflect the current state-of-the-art.

Evaluation of Criterion (5)

Section 6 of the ISV Plan discusses ISV performance measurement. Cognitive workload is measured using the [ ]. This approach acceptably meets the staff's review criterion.

(6) Anthropometric and Physiological Factors— Anthropometric and physiological factors include such concerns as visibility of indications, accessibility of control devices, and ease of control device manipulation that should be measured where appropriate. Attention should be focused on those aspects of the design that can only be addressed during testing of the integrated system, e.g., the ability of personnel to effectively use the various controls, displays, workstations, or consoles in an integrated manner.

Evaluation of Criterion (6)

Section 6 of the ISV Plan discusses ISV performance measurement. Information on general aspects of anthropometrics, including control room layout and workstation configuration, have been included in the [ ]. An assessment of anthropometric and physiological factors will also be made during HFE Design Verification. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.5.3 Performance Criteria

(1) Criteria should be established for the performance measures used in the evaluations. The specific criteria that are used for decisions as to whether the design is validated or not should be specified and distinguished from those being used to better understand the results.

Evaluation of Criterion (1)

Section 6.3, "Criteria," of the ISV Plan discusses the criteria to be used in evaluating performance measures. P/F performance measures are used to validate the design as was discussed in Section 4.5.2, "Performance Measure Selection, of this report. The general acceptance criteria are (1) [

].

For diagnostic measures, criteria are identified in ISV Plan, Table 6.3-2. The table provides criteria for all diagnostic measures. For example, the criteria for evaluating workload include: (1) average rating of workload across subjects from questionnaire is < 85 (range 0 to 100); (2) subjects demonstrate behavior, as specified in the scenario description for each scenario, that their workload is within a reasonable range and there are no indications of stress caused by

excessive workload; and (3) No workload issues are identified through questionnaire comments, debriefing, video and audio recording review. Failure to meet the criteria is evaluated by the HED resolution process (APP-OCS-GEH-420) to determine its priority.

This approach acceptably meets the staff's review criterion.

(2) The basis for criteria should be defined, e.g., requirement-referenced, benchmark referenced, normative referenced, and expert-judgment referenced.

Evaluation of Criterion (2)

Section 6.3, "Criteria," of the ISV Plan discusses the criteria to be used in evaluating performance measures. The basis for criteria for P/F measures is [ ]. The criteria established for diagnostic measures are based [ ]. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6 Test Design

18.11.8.2.4.6.1 Coupling Crews and Scenarios

(1) Scenario Assignment—Important characteristics of scenarios should be balanced across crews. Random assignment of scenarios to crews is not recommended. The value of using random assignment to control bias is only effective when the number of crews is quite large. Instead, the validation team should attempt to provide each crew with a similar and representative range of scenarios.

Evaluation of Criterion (1)

Section 3.2 of the ISV Plan discussed assignment of participants to trials. A final run order will be identified after pilot testing so aspects such as scenario duration can be determined. However, an example is provided in Table 3.3-1 of a counter balanced presentation of scenarios to crews. In the example, [ ]. Assignments are made [ ]. The constraints and considerations are clearly identified in the ISV plan. This approach acceptably meets the staff's review criterion.

(2) Scenario Sequencing—The order of presentation of scenario types to crews should be carefully balanced to provide reasonable assurance that the same types of scenarios are not always being presented in the same linear position, e.g., the easy scenarios are not always presented first.

Evaluation of Criterion (2)

Section 3.2 of the ISV Plan discussed assignment of participants to trials. As noted in the evaluation of Criterion (1), the final trial orders will be determined following pilot testing. One of the principles to be followed in that determining the final run order is to balance the order to accommodate the types of concerns raised in the review criterion. For example, the Plan states that "[ ]." Such considerations should minimize the possibility of linear position effects. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6.2 Test Procedures

(1) Detailed, clear, and objective procedures should be available to govern the conduct of the tests. These procedures should include:

- The identification of which crews receive which scenarios and the order that the scenarios should be presented.

- Detailed and standardized instructions for briefing the participants. The type of instructions given to participants can affect their performance on a task. This source of bias can be minimized by developing standard instructions.

- Specific criteria for the conduct of specific scenarios, such as when to start and stop scenarios, when events such as faults are introduced, and other information discussed in Section 11.4.3.2.4, "Scenario Definition."

- Scripted responses for test personnel who will be acting as plant personnel during test scenarios. To the greatest extent possible, responses to communications from operator participants to test personnel (serving as surrogate for personnel outside the control room personnel) should be prepared. There are limits to the ability to preplan communications since personnel may ask questions or make requests that were not anticipated. However, efforts should be made to detail what information personnel outside the control room can provide, and script the responses to likely questions.

- Guidance on when and how to interact with participants when simulator or testing difficulties occur. Even when a high-fidelity simulator is used, the participants may encounter artifacts of the test environment that detract from the performance for tasks that are the focus of the evaluation. Guidance should be available to the test conductors to help resolve such conditions.

- Instructions regarding when and how to collect and store data. These instructions should identify which data are to be recorded by:

  - simulation computers

  - special purpose data collection devices (such as situation awareness data collection, workload measurement, or physiological measures)

  - video recorders (locations and views)

  - test personnel (such as observation checklists)

  - subjective rating scales and questionnaires.

- Procedures for documentation, i.e., identifying and maintaining test record files including crew and scenario details, data collected, and test conductor logs. These instructions should detail the types of information that should be logged (e.g., when tests were performed, deviations from test procedures, and any

unusual events that may be of importance to understanding how a test was run or interpreting test results) and when it should be recorded.

Evaluation of Criterion (1)

The evaluation below is numbered to correspond to the bulleted criteria above.

1. Section 3.2 of the ISV Plan addresses crew assignment to scenarios. See the discussion of crew assignments in Section 4.6.1, "Coupling Crews and Scenario," Criterion 1 of this report above. This acceptably meets the subcriterion.

2. Section 5.2 of the ISV Plan addresses the requirements for crew briefing in general. The briefing will [ ]. Detailed information on crew briefings is included in the scenario descriptions for Scenarios 1, 2 and 12 in the ISV Scenario Plan. The information provided in these three detailed scenarios is complete and consistent with the high-level guidance in the ISV Plan. For example, the briefing for Scenario 2 is:

   [ ].

   This acceptably meets the subcriterion.

3. [ ]:

   - [ ]
   - [ ]
   - [ ]

   This acceptably meets the subcriterion.

4. Section 5.2.2, "Communications with ISV Personnel," of the ISV Plan describes the general approach for communicating with ISV crews. The Plan indicates that scripted responses will be used when test personnel act as plant personnel, such as a local operator. The Plan further states that "[ ]." These scripted responses are included in the ISV Scenario Plan. For example, for Scenario 1, the following instruction is provided:

   [ ].

   This acceptably meets the subcriterion.

5. Section 5.2.3, "Unforeseen Events," of the ISV Plan provides guidance on interacting with participants when unexpected difficulties arise. The guidance addresses events unrelated to the testing, such as fire drills, as well as related events, such as simulator anomalies. The plan outlines responsibilities for interacting with crews and guidance on resuming versus restarting trials. This acceptably meets the subcriterion.

6. Section 5.2.1, "General Procedures and Documentation," and Section 5.2.4, "Storage of Data," of the ISV Plan define the responsibilities and procedures for management of ISV

data. For example, the Plan identifies the ISV coordinator as the individual responsible for data management. With regard to simulator recorded data, the Plan indicates that "The discrete event data and plant parameter data from the simulator will be stored on a server and burnt onto discs. The file names for this data will identify the scenario number, the crew, and will be dated and time-stamped."

Further, the ISV Plan indicates that "At the end of each scenario, the ISV Coordinator will distribute and collect the completed post-trial questionnaires for the subjects, and at the end of the crews and observers participation in ISV, the ISV Coordinator will distribute and collect the final questionnaires for the subjects and observers. All of this information is hardcopy, and will be clearly marked and stored in a secure location."

This acceptably meets the subcriterion.

7. Section 5.2, "ISV Procedures," of the ISV Plan provides procedures for documenting all data collected during the ISV. The guidance in the ISV Plan addresses all forms of data, e.g., simulator logs and questionnaires. The procedures are sufficiently explicit to ensure data is not mishandled or lost (see examples in the evaluations above). This acceptably meets the subcriterion.

In summary, the ISV Plan provides detailed, clear, and objective procedures to govern the conduct of the ISV tests. This approach acceptably meets the staff's review criterion.

(2) Where possible, test procedures should minimize the opportunity of tester expectancy bias or participant response bias.

Evaluation of Criterion (2)

The ISV Plan indicates that observers will be independent of the project and that their assignment to trials will be systematically varied. The use of standardized and scripted responses should also help to minimize bias. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6.3 Test Personnel Training

(1) Test administration personnel should receive training on:

- the use and importance of test procedures
- experimenter bias and the types of errors that may be introduced into test data through the failure of test conductors to accurately follow test procedures or interact properly with participants
- the importance of accurately documenting problems that arise in the course of testing, even if due to test conductor oversight or error.

Evaluation of Criterion (1)

Section 4.3 of the ISV Plan addresses training of test personnel. It states that the training for the ISV staff will be sufficient to ensure effective execution of the test scenarios and data collection. This training will occur during the pilot testing of the simulator and ISV scenarios.

The training will be specific for the tasks to be performed during ISV. Test conductor roles will be rehearsed prior to ISV. The training will include how and when to communicate with the participants. Scripted responses will be provided for the operations support staff to perform specified roles as plant personnel in applicable scenarios (e.g., local operators). In addition, training will be given on the importance of the ISV procedures and the possible impact of not following the ISV procedures. This will help ensure consistency of the ISV staff performance and behavior across the scenarios. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6.4 Participant Training

(1) Participant training should be of high fidelity; i.e., highly similar to that which plant personnel will receive in an actual plant. The participants should be trained to provide reasonable assurance that their knowledge of plant design, plant operations, and use of the HSIs and procedures is representative of experienced plant personnel. Participants should not be trained specifically to perform the validation scenarios.

Evaluation of Criterion (1)

Sections 4.1 and 4.1.3 of the ISV Plan discuss participant training. Operations personnel from the customer utilities participating in the AP1000 training program will be used for ISV. Training will include both classroom and hands-on simulator components. The training will be developed and delivered by the Westinghouse Training Group.

The training program will provide personnel with detailed AP1000 systems and plant knowledge. The program will be presented using a combination of classroom instruction, self-study, procedure walk through, and exercises. Training will include [ ] of AP1000 systems training and [ ] of procedure/simulator based training. This will include EOPs, AOPs, and GOPs.

This approach acceptably meets the staff's review criterion.

(2) Participants should be trained to near asymptotic performance (i.e., stable, not significantly changing from trial to trial) and tested prior to conducting actual validation trials. Performance criteria should be similar to that which will be applied to actual plant personnel.

Evaluation of Criterion (2)

Sections 4.1 and 4.1.3 of the ISV Plan describe the training program for ISV participants. As discussed under Criterion (1), the program provides sufficient training such that the skill and knowledge levels of participants should not be significantly changing between trials. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6.5 Pilot Testing

(1) A pilot study should be conducted prior to conducting the integrated validation tests to provide an opportunity to assess the adequacy of the test design, performance measures, and data collection methods.

Evaluation of Criterion (1)

Section 3.3 of the ISV Plan indicates pilot testing will be performed to address these aspects of the test. The objectives of simulator pilot testing are to demonstrate that the simulator responds in a manner similar to the reference unit while utilizing the plant operating procedures, to ensure simulator readiness, and to minimize the likelihood of test failures or delays. The pilot testing will be carried out in [ ], as described in the ISV Scenario Plan, Appendix C. In addition to the testing of the simulator model, thorough pilot testing of all scenarios will be carried out. This approach acceptably meets the staff's review criterion.

(2) If possible, participants who will operate the integrated system in the validation tests should not be used in the pilot study. If the pilot study must be conducted using the validation test participants, then:

- the scenarios used for the pilot study should be different from those used in the validation tests; and
- care should be given to provide reasonable assurance that the participants do not become so familiar with the data collection process that it may result in response bias.

Evaluation of Criterion (2)

Section 3.3 of the ISV Plan states that the participants for ISV will not be involved in pilot testing. Rather, the pilot testing will be performed by the Westinghouse simulator development staff, with support as needed from other Westinghouse personnel. This acceptably addresses Criterion (2). This approach acceptably meets the staff's review criterion.

18.11.8.2.4.7 Data Analysis and Interpretation

(1) Validation test data should be analyzed through a combination of quantitative and qualitative methods. The relationship between observed performance data and the established performance criteria should be clearly established and justified based upon the analyses performed.

Evaluation of Criterion (1)

Section 7, "Processing of Results," of the ISV Plan [ ].
This approach acceptably meets the staff's review criterion.

(2) For performance measures used as pass/fail indicators, failed indicators must be resolved before the design can be validated. Where performance does not meet criteria for the other performance measures, the results should be evaluated using the HED evaluation process.

Evaluation of Criterion (2)

Section 7.3 of the ISV Plan indicates that each scenario is run [ ] times with [ ] different crews. If a scenario fails a P/F criterion an HED is defined and resolved. The scenario is then rerun a minimum of [ ] times with [ ]. With respect to diagnostic measures, observation of a small number of HEDs will result in a [ ] trial being run using a [ ]. This

approach will help confirm whether an HED exists or not. The applicant indicated that, if the results of the [ ] trial confirm an issue, an HED will be identified and resolved. The scenario then will be re-run [ ] times using [ ]. This approach acceptably meets the staff's review criterion.

(3) The degree of convergent validity should be evaluated, i.e., the convergence or consistency of the measures of performance.

Evaluation of Criterion (3)

Section 7.2 of the ISV Plan indicates that the degree of convergence of measures will be assessed in the interpretation of the results. The Plan states:

> The degree to which convergent (i.e., consistent) results are observed from different measurement techniques will be [ ] and the results will presented in the ISV results report. The analysis will determine if the different measurement techniques indicate the same problems. If so, it strengthens the conclusion that a problem exists and it needs to be addressed. Likewise, if none of the measurement techniques indicates that there is a problem (i.e., different measurement techniques record successful performance), then it increases the degree of certainty that a problem does not exist.

This approach acceptably meets the staff's review criterion.

(4) The data analyses should be independently verified for correctness of analysis.

Evaluation of Criterion (4)

Section 7.2 of the ISV Plan indicates that independent verification of results will be performed using applicable Westinghouse quality assurance procedures provided in APP-GW-GAP-100, "Inter-Business Unit Edition Policies & Procedures." This approach acceptably meets the staff's review criterion.

(5) The inference from observed performance to estimated real-world performance should allow for margin of error; i.e., some allowance should be made to reflect the fact that actual performance may be slightly more variable than observed validation test performance.

Evaluation of Criterion (5)

Section 6.3.1, "Pass/Fail Criteria," of the ISV Plan indicates that for P/F criteria, each scenario specifies that [ ]. In order to ensure margin, there are also typically acceptance criteria relating to not exceeding the [ ].

Regarding the risk-important HAs, [ ]. Section 6.3.1 of the ISV Plan states that "In a number of cases in the PRA, the [ ]. Therefore, the [ ] to perform the risk-important human actions will be closely monitored. If a case occurs where the [ ] is potentially insufficient to ensure reliable operator performance, this will be identified as [ ]." The method of identifying and documenting the [ ] for performing the risk-important HAs is described in the ISV Scenario Plan, in the Scenario Specifications and in the Observer Guides for the scenarios.

RAI-SRP18-COLP46 requested more information on the mechanism for specifying and documenting the [ ] needed to accomplish the risk-important HAs in the ISV Scenarios. In its response, dated August 2, 2010, the applicant stated that the "[ ]. The HFE analyst will use [ ] in Section 6.2, "Methods," of the ISV Plan to complete this calculation. This approach limits the potential of results being influenced by the observer's judgment or the observer missing a task step, event or operator action. The [ ] provide an objective confirmation of the observation results.

This approach acceptably meets the staff's review criterion.

18.11.8.2.4.8 Validation Conclusions

(1) The statistical and logical bases for determining that performance of the integrated system is and will be acceptable should be clearly documented.

Evaluation of Criterion (1)

Section 7.4 of the ISV Plan provides the commitment to document the basis for validation conclusions. The ISV Plan states "The basis for concluding that the AP1000 MCR, HSI resources, procedures, and operator training are adequate (or not) will be described (i.e., that the integrated system performed acceptably during testing and can be expected to support safe operation in actual use)." This approach acceptably meets the staff's review criterion.

(2) Validation limitations should be considered in terms of identifying their possible effects on validation conclusions and impact on design implementation. These include:

- aspects of the tests that were not well controlled
- potential differences between the test situation and actual operations, such as absence of productivity-safety conflicts
- potential differences between the validated design and plant as built (if validation is directed to an actual plant under construction where such information is available or a new design using validation results of a predecessor).

Evaluation of Criterion (2)

Section 7.4 of the ISV Plan provides the commitment to document test limitations. This approach acceptably meets the staff's review criterion.

18.11.8.3 Conclusion

The staff concludes that APP-OCS-GEH-320 and APP-OCS-GEH-321 provide implementation plans that conform to the NUREG-0711 criteria associated with ISV. The staff's review of the AP1000 ISV Plan and the ISV Scenario Plan concludes that the plans are comprehensive and thorough and provide reasonable assurance that the ISV will effectively identify any operator challenges associated with the HSI design. Open Item OI-SRP18-COLP-03A was created to track the completion of these documents. Based on the preceding information, the staff

concludes that this open item and the corresponding element of ITAAC Design Commitment 4 (DCD, Revision 15) are complete and closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.11.9 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 4 of 5, Issue Resolution Verification

18.11.9.1 Summary of Technical Information

ITAAC Design Commitment 4 reads as follows:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (part 4): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the ...Issue Resolution Verification activity.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.11.9.2 Evaluation

NUREG-0711, Section 11.4.4.2, Criterion 1, states the following:

> Discrepancies could be acceptable within the context of the fully integrated design. If sufficient justification exists, a deviation from the guidelines may not constitute an HED. The technical basis for such a determination could include an analysis of recent literature or current practices, tradeoff studies, or design engineering evaluations and data. The applicant should identify unjustified discrepancies as HEDs to be addressed by the HED resolution.

Evaluation of Criterion 1

Each of the three V&V implementation plans previously referenced (APP-OCS-GEH-120, APP-OCS-GEH-220, and APP-OCS-GEH-320) include sections directing that all discrepancies be documented on a discrepancy worksheet. The Operation and Control System product manager screens the worksheet. If the discrepancy can be directly justified, it is not considered an HED. Unjustified discrepancies are identified as HEDs, and the applicant must address them using the formal resolution process in APP-OCS-GEH-420. The staff concludes that procedures referenced in this section provide sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for HED justification.

NUREG-0711, Section 11.4.4.2, Criterion 2, states the following:

The HED analysis should include the following:

- Plant system—The potential effects of all HEDs relevant to a single-plant system should be evaluated. The potential effects of these HEDs on plant safety and personnel performance should be determined, in part, by the safety significance of the plant system, their effect on the accident analyses summarized in the SAR, and their relationship to risk-significant sequences in the plant PRA.

- HED scope—The scope of the HED should consider the following:

  - Global features HEDs—These HEDs relate to configurational and environmental aspects of the design, such as lighting, ventilation, and traffic flow. They relate to general human performance issues.

  - Standardized features HEDs—These HEDs relate to design features that are governed by the applicant's design guidelines used across various controls and displays of the HSI (e.g., display screen organization and conventions for format, coding, and labeling). Because a single guideline may be used across many aspects of the design, a single HED could be applicable to many personnel tasks and plant systems.

  - Detailed features HEDs—These HEDs relate to design features that are not standardized, thus their generality has to be assessed.

  - Other—This subcategory specifically pertains to HEDs identified from integrated system validation that cannot be easily assigned to any of the three preceding categories.

- Individual HSI or procedure—HEDs should be analyzed with respect to individual HSIs and procedures. The potential effects of these HEDs on plant safety and personnel performance are determined, in part, by the safety significance of the plant system that is related to the particular component.

- Personnel function—HEDs should be analyzed with respect to individual personnel functions. The potential effects of these HEDs is determined, in part, by the importance of the personnel function to plant safety (e.g., consequences of failure) and the cumulative effect on personnel performance (e.g., degree of impairment and types of potential errors).

  The applicant should also analyze HEDs with respect to the cumulative effects of multiple HEDs on plant safety and personnel performance.

  In addition to addressing the specific HEDs, the analysis should treat the HEDs as indications of potentially broader problems.

Evaluation of Criterion 2

In APP-OCS-GEH-420, Section 2.2, the applicant stated that the [ ], which the OCS product manager approves. The assessment [ ]. This prioritization addresses the first bullet from the NUREG-0711 criterion above:

- Priority 1: [ ]
  - [ ].
  - [ ].
- Priority 2: [ ]:
  - [ ]
  - [ ]
  - [ ]
- Priority 3: All others

APP-OCS-GEH-420, Section 2.3, addresses the HED scope. In this section, the applicant stated that the cumulative effects of Priority 1 and 2 HEDs are analyzed by organizing HEDs into the following categories:

- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]

This last category identifies [ ], and others using the same definitions as the NUREG criterion. By separating the [ ]. This addresses the remaining parts of the NUREG-0711 criterion above.

APP-OCS-GEH-420, Section 2.5, states that an [ ]; an evaluation is performed to [ ]. The categorization described above is used to identify these generic implications.

The staff concludes that APP-OCS-GEH-420 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for HED analysis.

NUREG-0711, Section 11.4.4.2, Criterion 3, states the following:

> The applicant should use a systematic evaluation to identify HEDs for correction. Priority 1 HEDs are those with direct, indirect, or potential safety consequences. Priority 2 HEDs are those that do not have significant safety consequences, but do have potential consequences to plant performance/operability, nonsafety-related personnel performance/efficiency, or other factors affecting overall plant operability. The remaining HEDs are those that do not satisfy the criteria associated with the first and second priorities. Resolution of these HEDs

> is not an NRC safety concern but may be resolved at the discretion of the applicant.

Evaluation of Criterion 3

This criterion is addressed in the previous section.

NUREG-0711, Section 11.4.4.2, Criterion 4, states the following:

> The applicant should fully document each HED, including assessment category (priority for correction), associated plant system, associated personnel function, and associated HSI or procedure. The documentation should clearly show whether the HED was dismissed or identified as needing design modification, and the basis for this determination in terms of consequence to plant safety or operation should be clearly described.

Evaluation of Criterion 4

Section 2 of APP-OCS-GEH-420 includes documentation requirements. In summary, when the HFE engineer justifies an HED, he or she [ ]. When an HED is resolved by a design solution, a [ ] is used in conjunction with a [ ] to identify the best solution. Solutions will be consistent with the system requirements used to design the system. Design solutions follow the design process which documents the impact on safety.

The staff concludes that APP-OCS-GEH-420 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for HED evaluation documentation.

NUREG-0711, Section 11.4.4.2, Criterion 5, states the following:

> The applicant should identify design solutions to correct HEDs. The design solutions should be consistent with system and personnel requirements identified in the preparatory analysis (i.e., operating experience review, function and task analysis, and HSI characterization).

Evaluation of Criterion 5

APP-OCS-GEH-420, Section 2.6, states that the design solution will be consistent with the system requirements used to design the system. By comparing proposed changes to the HFE design to the original system requirements, the applicant will ensure the original design basis is maintained or adjusted as necessary.

The staff concludes that APP-OCS-GEH-420 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for development of HED design solutions.

NUREG-0711, Section 11.4.4.2, Criterion 6, states the following:

> The applicant should evaluate designs by repeating the appropriate V&V analyses. When the problems identified by an HED cannot be fully corrected, the applicant should provide appropriate justification.

Evaluation of Criterion 6

In APP-OCS-GEH-420, Section 2.8, the applicant stated that, for design solutions associated with the HFE design verification plan or HFE task support verification plan, independent verifiers will evaluate the HSI design changes using the same standards, guidance, and methodology as described in the applicable verification plan. For design solutions associated with the integrated system validation plan, the human factors team will determine the appropriate evaluation process, using a graded approach, based on the complexity and impact of the design changes. Independent verifiers will then perform the evaluation process.

The staff concludes that APP-OCS-GEH-420 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for design solution evaluation.

18.11.9.3 Conclusion

The staff concludes that APP-OCS-GEH-420 provides an implementation plan that satisfactorily addresses the NUREG-0711 criteria associated with tracking and resolving HEDs. This implementation plan provides reasonable assurance that issues will be identified during all stages of the design process and that these issues will be prioritized and resolved in an efficient manner. This element of ITAAC Design Commitment 4 (DCD Revision 15) as described above is complete and closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.11.10 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 5 of 5, Plant HFE/HSI (as Designed at the Time of Plant Startup) Verification

18.11.10.1 Summary of Technical Information

ITAAC Design Commitment 4 reads as follows:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (part 5): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the.... Plant HFE/HSI (as designed at the time of plant startup) Verification activity.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.11.10.2 Evaluation

The applicant submitted APP-OCS-GEH-520 to address this part of ITAAC 4. Open Item OI-SRP18-COLP-4A was created to track completion of the staff's review of this document. The acceptance criteria for this implementation plan are found in NUREG-0711 Section 12, "Design Implementation." The staff evaluation of this plan is being provided in this section of the SER so that material applicable to ITAAC 4 closure is kept together.

NUREG-0711, Section 12.4.6, Criterion 1, states the following:

> Aspects of the design that were not addressed in V&V should be evaluated using an appropriate V&V method. Aspects of the design addressed by this criterion may include design characteristics such as new or modified displays for plant-specific design features and features that cannot be evaluated in a simulator such as Control Room lighting and noise.

Evaluation of Criterion 1

In APP-OCS-GEH-520, Section 1, the applicant states that specific aspects of the OCS. HSI design that cannot be evaluated in a simulator will be evaluated via a walk down of the applicable plant area after construction. This plan applies to all control areas included in the HFE scope including the MCR, the remote shutdown room, TSC, radioactive waste control, and LCSs. APP-OCS-GER 120 and APP-OCS-GER-220 document the results of the design and task analysis verification and are used as the basis for identifying design verifications that have not been completed. The procedure specifically identifies lighting, noise, ambient temperature and humidity, the closed circuit TV system, communication facilities, and maintainability as areas that will be evaluated. The applicant indicates that where appropriate physical measurements will be taken for key environmental features including lighting, thermal conditions, and acoustics.

The staff concludes that APP-OCS-GEH-520 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion.

NUREG-0711, Section 12.4.6, Criterion 2, states the following:

> The final (as-built in the plant) HSIs, procedures, and training should be compared with the detailed design description to verify that they conform to the design that resulted from the HFE design process and V&V activities. Any identified discrepancies should be corrected or justified.

Evaluation of Criterion 2

In APP-OCS-GEH-520, the applicant states that the as-built HSIs will be verified to the same as those that resulted from the HFE program. A team is used to complete the verification. The team uses the expected design configuration and control information, including the style guide, detailed design descriptions, and guidance on evaluating maintainability, to compare the as-built design against. The adequacy of procedures and training are addressed as part of the staff's operating program inspections.

The staff concludes that APP-OCS-GEH-520 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion.

NUREG-0711, Section 12.4.6, Criterion 3, states the following:

> All HFE-related issues documented in the issue tracking system should be verified as adequately addressed.

Evaluation of Criterion 3

In APP-OCS-GEH-520, the applicant states that all HEDs will be verified as being adequately addressed.

The staff concludes that APP-OCS-GEH-520 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion.

18.11.10.3 Conclusion

The staff concludes that APP-OCS-GEH-520 provides an implementation plan that satisfactorily addresses the NUREG-0711 criteria associated with the as-built design verification. The scope and methods described provide reasonable assurance that the as-built HFE design configuration will mirror the design described in the DC. This element of ITAAC Design Commitment 4 (DCD Revision 15) as described above is complete and closed; therefore, Open Item OI-SRP18-COLP-04A is closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.16 Tier 2* Information

The staff has determined that the following information referenced in DCD Tier 2, Chapter 18, Revision 17, must be designated as Tier 2* information in the AP1000 DCD. This information is in addition to the information identified as Tier 2* for Revision 15 of the DCD as documented in Section 18.16 of NUREG-1793.

1. APP-OCS-GEH-120, Revision B, "AP1000 Human Factors Engineering Design Verification Plan." (This report explains the applicant's method for design verification.)

2. APP-OCS-GEH-220, Revision B, "AP1000 Human Factors Engineering Task Support Verification Plan." (This report explains the applicant's method for task support verification.)

3. APP-OCS-GEH-320, Revision D, "AP1000 Human Factors Engineering Integrated System Validation Plan." (This report explains the applicant's method for performing the ISV.)

4. APP-OCS-GEH-420, Revision B, "Human Factors Engineering Discrepancy Resolution Process." (This report explains the applicant's method for resolving HEDs.)

5. APP-OCS-GEH-520, Revision B, "AP1000 Plant Startup HFE Design Verification Plan." (This report explains the applicant's method for verifying the as-built design.)

Based on guidance provided in the NUREG-0800 Section 14.3 and Branch Technical Position HICB-16 (Guidance on the level of detail required for design certification application under 10 CFR Part 52, "License, certifications, and approvals for nuclear power plants") the staff has concluded that all Tier 2* material associated with Chapter 18 will revert to Tier 2 after the plant first achieves full-power operation. This includes Tier 2* information identified in NUREG-1793. This is a change from how HFE-related Tier 2* material was addressed for the previously approved DCD Revision 15 where there was no expiration date. 10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design," Sections VIII.B.6.b and VIII.B.6.c describe regulatory requirements associated with Tier 2* material and will be amended to reflect this change.

The additional documents were identified as Tier 2* because they describe the specific process the applicant will use to accomplish the final HFE design. The staff has verified that this process conforms to the regulatory guidance in NUREG-0737, which in turn supports the staff's conclusions that the HFE design provides reasonable assurance the Control Room staff can safely control plant operations via the HSIs. These documents also include acceptance criteria that will be used when inspecting the final HFE design conforms to Table 3.2-1 ITAAC.

The applicant responded to RAI-SRP18-COLP-23 R3, dated August 2, 2010, revising Tier 1 and Tier 2 of the DCD to reflect these additional Tier 2* references. In addition, the introduction of the DCD addresses Tier 2* references and specifies when the * designation expires in Table 1-1. The applicant responded to RAI-SRP18-COLP-54, dated August 18, 2010, revising the table to include the additional references listed above and to note all HFE Tier 2* references will expire when the COL holder first achieves 100 percent power operation. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19. SEVERE ACCIDENTS

19.0 Background

In December 2005, the U.S. Nuclear Regulatory Commission (NRC) issued Supplement 1 to NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." NUREG-1793 and its supplement documented the basis for certifying the AP1000 design. Subsequently, the agency has issued new or revised requirements and guidance for addressing severe accidents in the following documents:

Title 10 of the Code of Federal Regulations (10 CFR) Part 52

The Commission issued 10 CFR Part 52, ""Licenses, certifications, and approvals for nuclear power plants," on April 18, 1989. This rule provides for issuing early site permits, standard design certifications (DCs), and combined licenses (COLs) with conditions for nuclear power reactors. It details the review procedures and licensing requirements for applications for these new permits, certifications, and licenses. It is intended to achieve the early resolution of licensing issues, as well as to enhance the safety and reliability of nuclear power plants.

The NRC revised the rule on August 28, 2007. Specifically, 10 CFR 52.47, "Contents of applications; technical information," now requires an application for a DC to describe the design-specific probabilistic risk assessment (PRA) and its results. 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," now requires each COL applicant to describe the plant-specific PRA and its results.

Regulatory Guide 1.200

Regulatory guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,", was issued in February 2004, then revised in January 2007 and March 2009. It describes one acceptable approach for determining whether the quality of PRA provides sufficient confidence in the results to support regulatory decision making for light-water reactors. RG 1.200 endorses, with certain restrictions, a standard published by the American Society of Mechanical Engineers (ASME); ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," including Addenda A and B. It also endorses Nuclear Energy Institute (NEI) 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance."

Interim Staff Guidance

DC/COL-ISG-1, "Interim Staff Guidance on Seismic Issues of High Frequency Ground Motion in Design Certification and Combined License Applications," which clarified the implementation of a performance-based approach for determining site-specific ground motion and methodology for evaluating the effects of high frequency ground motion.

DC/COL-ISG-3, "Probabilistic Risk Assessment Information to Support Design Certification and Combined License Applications," clarifies the expectations of the staff with respect to the level of detail to be described and results to be reported in applications.

DC/COL-ISG-20, "Interim Staff Guidance on Implementation of a Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment," provides more detail on the seismic margin analysis and identifies what to document at the time of application for a COL, and prior to operation.

19.1 Probabilistic Risk Assessment

19.1.1 Introduction

Westinghouse Electric Company, LLC (Westinghouse or the applicant) filed an application for an amendment to the AP1000 DC rule (10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design") and provided a revised design control document (DCD). Westinghouse had submitted a design-specific PRA of the AP1000 design as part of the AP1000 design documentation for the certified design. Westinghouse did not submit a revised PRA report with the amendment request; however, Westinghouse did describe, in a number of technical reports (TRs), the changes to the PRA that would result from the design modifications proposed in the amendment. The proposed changes to the DCD reflect these modifications.

The NRC's regulations at 10 CFR Part 52 no longer require submittal of the PRA report. Instead, applicants are to provide, in the DCD, a description of the PRA and a summary of its results. The design-specific PRA is available for the staff's review and still forms the basis for the site-specific, plant-specific PRAs that COL applicants must describe and COL licensees must upgrade and update before loading fuel.

Since certification of the AP1000 design, Westinghouse upgraded the PRA as part of a conversion from proprietary software to a widely used program, the Computer-Aided Fault-Tree Analysis System (CAFTA). The PRA was also updated to reflect proposed design changes. As part of the AP1000 DC amendment application, it reported all resulting changes to the insights, assumptions, and results of the analysis.

In addition to the revised DCD, the staff reviewed the following AP1000 COL standard TRs:

- Westinghouse Commercial Atomic Power (WCAP)-16555, APP-GW-GL-011, "AP1000 Identification of Critical Human Actions and Risk Important Tasks," March 2006
- TR-6, APP-GW-GLR-021, "AP1000 As-built COL Information Items," June 2006
- TR-34, APP-GW-GLN-016, "AP1000 Licensing Design Change Document for Generic Reactor Coolant Pump," November 2006
- TR-36, APP-GW-GLR-016, "AP1000 Pressurizer Design," May 2006
- TR-66, APP-GW-GLR-070, "Development of Severe Accident Management Guidance," January 2007
- TR-88, APP-GW-GLR-065, "AP1000 Instrumentation & Control (I&C) Data Communication and Manual Control of Safety Systems and Components," Revision 1, May 2009

- TR-97, APP-GW-GLN-022, Revision 1, "DAS [diverse actuation system] Platform Technology and Remote Indication Change," May 2007
- TR-101, APP-GW-GLR-101, "AP1000 Probabilistic Risk Assessment Site-Specific Considerations," Revision 1, October 2007
- TR-102, APP-GW-GLR-102, "AP1000 PRA Update Report," Revision 1, November 2009
- TR-105, APP-GW-GLN-105, Revision 2, "Building and Structure Configuration, Layout and General Arrangement Design Updates," October 2007
- TR-106, APP-GW-GLN-106, Revision 1, "Mechanical System and Component Design Update," September 2007
- TR-130, APP-GW-GLR-130, "Editorial Format Changes Related to Combined License Applicant and Combined License Information Items," June 2007
- TR-134, APP-GW-GLR-134, Revision 5, "AP1000 DCD Impacts to Support COLA Standardization," June 2008
- TR-135, APP-PRA-GER-001, "AP1000 Design Change Proposal Review for PRA and Severe Accident Impact," Revision 1, December 1, 2009
- TR-147, APP-GW-GLN-147, Revision 1, "AP1000 CR and IRWST Screen Design," March 2008

This information is generic to the design and applies to all COL applications that reference the AP1000 DC.

19.1.1.1 Background and NRC Review Objectives

The general objectives of the NRC's review of the most recent AP1000 DCD revision include the following:

- identification of new risk-informed safety insights based on systematic evaluations of risk associated with the amended design
- confirmation that regulatory treatment of nonsafety systems (RTNSS) remains appropriate
- confirmation that the DC requirements, such as inspection, tests, analyses, and acceptance criteria (ITAAC), design reliability assurance program (D-RAP), and technical specifications, as well as COL and interface requirements, are amended as appropriate
- confirmation that the conclusions reached in the previous certification remain valid

During the construction stage, the COL applicant will ensure that detailed design documents are consistent with the certified design so that the key assumptions and risk insights from the PRA

remain valid. (These assumptions and insights are documented in DCD Table 19.59-18. The D-RAP ITAAC confirm this by ensuring that appropriate quality controls have been applied in the development of detailed design for procurement and construction.) The COL applicant will ensure, through other ITAAC and preoperational programs, that the configuration of the plant, as built, is consistent with the detailed design. The Commission believes that updated PRA insights, if properly evaluated and used, could strengthen programs and activities in areas such as training, development of emergency operating procedures (EOPs), reliability assurance, maintenance, and evaluations performed pursuant to 10 CFR 50.59, "Changes, tests and experiments." The design-specific PRA, developed as part of the DC process, should be revised to account for site-specific information, as-built (plant-specific) information refinements in the level of design detail, technical specifications, plant-specific EOPs, and design changes. The COL licensee is responsible for these updates. This is part of COL Information Item 19.59.10-2.

The NRC requires the COL applicant to develop a plant-specific PRA based on the design-specific PRA. At the time of application, the plant-specific PRA of internal events (both at power and shutdown) must address, at a minimum, proposed deviations from the certified design. The plant-specific PRA of external events must evaluate external events applicable to the proposed site and confirm that they are bounded by the PRA. This is also part of COL Information Item 19.59.10-2. The staff expects that the COL applicant and licensee will use the plant-specific PRA and revised failure rates (when available) to update, as appropriate, its reliability assurance programs (including the quality assurance program and the maintenance rule program).

19.1.1.2 Evaluation of Probabilistic Risk Assessment Quality and Closure of Open Issues

The NRC staff evaluated the information submitted by the applicant in accordance with NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." For Chapter 19, the staff used Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," and Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

In TR-102, the applicant described conversion of the PRA modeling software package from its proprietary program, WesSAGE, to the more widely used program, CAFTA. The applicant also updated the AP1000 PRA to include the most recent instrumentation and controls (I&C) design information. The applicant documented the basis for its determination that structures, systems, and components (SSCs) modeled in the PRA were not affected by other design changes in a manner that affected the PRA.

In its original review of the AP1000 PRA, the staff relied on the similarity between the AP600 and AP1000 certified designs to reduce the review effort. This similarity (e.g., in system design and overall plant layout) allowed the use of the AP600 PRA as the starting point in the development of the AP1000 PRA. Similarly, the PRA associated with the currently certified AP1000 design was the starting point for upgrading and updating the AP1000 PRA in support of the DC amendment. In addition to reviewing the description of changes to the PRA, the staff reviewed the description of the new I&C design, specifically the plant control system (PLS) and the protection and safety monitoring system (PMS).

The staff used reported PRA results, as well as the results of sensitivity, uncertainty, and importance analyses, to focus its review. The staff also used applicable insights from previous PRA studies regarding key parameters and design features.

The review of the quality and completeness of the AP1000 PRA included the issuance of several requests for additional information (RAIs) to the applicant related to TR-102. Following its review of the responses to the RAIs, in August 2007, the staff conducted an audit at the applicant's offices, focusing on three principal areas:

1. assessment of the applicant's process to upgrade the PRA model, including the process by which they assessed the design and operational changes for potential impact on the PRA

2. review of the changes to the model since the applicant submitted the previous PRA report, especially those resulting from proposed changes to the certified design

3. inspection of the model itself to confirm that the model accurately reflected modifications to the design and that the model is a suitable basis for PRAs required of COL applicants that reference the AP1000 DC

In the process, the staff reviewed the qualifications of personnel involved in PRA-related activities and found them to be acceptable. The staff also examined the procedures the applicant used to review modifications for their potential impact on the PRA, to identify and correct problems in the model, to implement changes to the model, and to assess the results of analysis. During this review, the staff developed further requests for additional information based on NUREG-0800 Chapter 19, discussed below.

Section 19.1.10 of this report provides a summary and the resolution of open items. These are issues resulting from the review of Chapter 19 that had not been resolved when the draft of this report was prepared for review by the Advisory Committee on Reactor Safeguards.

19.1.2 Special Advanced Design Features

19.1.2.1 Special Advanced Design Features for Preventing Core Damage

The applicant proposed changes to the certified AP1000 design that have the potential to affect the PRA. The following sections discuss these changes.

19.1.2.1.2 Defense-In-Depth Active Nonsafety-Related Systems

The AP1000 design incorporates several active systems that are capable of performing some of the same functions as those performed by the safety-related passive systems. The availability of such redundant systems minimizes the challenge to the safety-related passive systems by providing core cooling during normal plant shutdowns and serving as a first line of defense during accidents.

The diverse actuation system (DAS) provides an alternate means for initiating automatic and manual reactor trip and actuation of selected engineered safety features that is diverse from the safety-related PMS. An additional DAS squib valve control cabinet, spatially separated from the DAS cabinet in the control room, provides additional confidence that operators can take manual actions to depressurize the reactor coolant system (RCS) and initiate key functions, such as

in-containment refueling water storage tank (IRWST) injection, containment recirculation, and IRWST drain to containment. In DCD Table 19.59-18, the applicant clarified the degree of diversity between the PMS and the DAS. Section 7.7 of this report includes the staff evaluation of this modification. In RAI-SRP19.0-SPLA-06, the staff requested information on the potential of this modification to affect the timing of steps taken to mitigate an anticipated transient without scram (ATWS) event (positively or negatively) and to reduce risk by providing a spatially diverse actuation station.

In a letter dated August 21, 2008, the applicant responded that the addition of the remote DAS cabinet would not result in a significant change to the risk importance of any SSC or human action. The probability of using this remote cabinet is very low, as it would require several very unlikely events to occur. An example of such a situation would be the need to use manual controls coupled with the need to evacuate the control room and the failure or unavailability of the remote PMS panels. It is very likely that a fire or event that would make the control room and the remote PMS panel unavailable for manual operation would also result in a successful automatic shutdown of the AP1000. In addition, human reliability analyses (HRAs) for the DAS use values that represent low probability of success, high-stress situations. This supports the conclusion that the modification would not result in a significant change to the risk importance of any SSC or human action. For that reason, the AP1000 PRA does not model the use of this remote panel separately from use of the one in the control room.

The staff noted that the PRA models manual action as a single basic event irrespective of the location from which that action is taken. Furthermore, the proposed modification reduces uncertainty in the performance of the action. Because the need to use the remote DAS cabinet requires multiple, simultaneous, and highly unlikely events, the risk importance of SSCs or human actions would not be significantly altered by additional detail in the model. For these reasons, the staff concludes that the applicant's decision not to alter the modeling of this system is conservative and acceptable. The staff considers RAI-SRP19.0-SPLA-06 resolved.

19.1.2.1.7 Redundant Long-Term Recirculation Systems

RCS recirculation is required for long-term core cooling during loss-of-coolant accidents (LOCAs) and whenever the feed-and-bleed method is used to cool the core during an accident. In the AP1000, recirculation can be achieved either by gravity (through the safety-related IRWST injection lines) or pumping (through the nonsafety-related normal residual heat removal system (RNS)) with suction from the containment sump. Two redundant recirculation lines exist (one for each of the two redundant IRWST injection lines). Furthermore, each recirculation line has two paths that are redundant, with the exception of the recirculation screens. Though there are two separate screens, the applicant does not characterize them as redundant and explicitly models their common-cause failures (CCFs). Section 6.2.1.8 of this report documents the staff's evaluation of the screen design.

In RAI-SRP19.0-SPLA-04, the staff asked the applicant to discuss the impact of changes to the design of the recirculation system on the results and insights of the shutdown risk assessment.

In a letter dated July 22, 2008, the applicant stated that the structural integrity of the new recirculation screen configuration exceeds that of the screen-like material typically used in current pressurized-water reactor (PWR) sump screens, precluding the need for trash racks. Screen testing demonstrated that the new screens will not experience a significant head loss while operating within design-basis flow/debris conditions. A qualitative assessment concluded

that the enhancement will reduce failure probability, while the increased flow area will improve accident response.

The staff finds it reasonable to expect that the proposed modification will improve performance of the recirculation screens as compared to the certified design. Although the large, interconnected screens are not independent, the staff finds that the applicant adequately addressed CCF of the screens. For these reasons, the staff concludes that the applicant's decision not to alter the modeling of this system is conservative and acceptable. The staff considers RAI-SRP19.0-SPLA-04 resolved.

19.1.2.1.9 Canned Reactor Coolant Pumps

The AP1000 design originally specified canned reactor coolant pumps (RCPs). In TR-34, the applicant specified sealless RCPs, which may be canned-motor or wet-winding pumps. For both canned-motor and wet-winding pumps, the motor and all rotating components are inside a pressure vessel. The pressure vessel consists of the pump casing, thermal barrier, stator shell, and stator cap, all of which are designed for full RCS pressure. Because the rotor and shaft connecting it to the impeller are contained within the pressure boundary, a seal is not required to restrict leakage out of the pump into containment. In addition, the heat exchanger that cools the RCP has been modified; it is now external to the pump. The applicant asserts that these changes do not alter the PRA model.

Section 5.4 of this report discusses the staff's evaluation of changes to the RCP design. The pump design is important because the use of sealless RCPs in the AP1000 design eliminates the RCP seal LOCA (an important contributor to risk for most operating commercial PWRs). In addition, water is used to lubricate and remove heat from pump bearings, eliminating the need for RCP lubricating oil systems and the attendant fire hazard. Because the proposed design alternative of a wet-winding rotor changes neither the failure modes of the RCP and its heat exchanger nor the estimated reliability of these components, the staff concludes that no change to the internal events PRA model is required. However, this change had an impact on the seismic margin analysis. The staff's evaluation of this issue is addressed in Section 19.1.5.1, "Probabilistic-Risk-Assessment–Based Seismic Margin Analysis."

19.1.2.1.10 Improved Control Room Design and Digital Instrumentation and Control Systems

The AP1000 control room is an advanced design that is expected to provide information that is presented to the operator in a way that is more easily used than the displays in currently operating reactor designs. Similarly, control is expected to be easy and consistent, and nearly all actions can be performed from a single station. Section 7.1.4 of this report documents the staff's review of the AP1000 control room design.

The PRA took no credit for the impact of the advanced control room on normal operations and emergency response (e.g., initiating event frequency or HRA). Because the impact of an advanced control room is still the subject of research and control room design verification and validation cannot be performed until a control room is simulated, the staff concludes that this approach is conservative and acceptable for the DC.

During the August 2007 audit of the applicant's PRA, the staff identified a discrepancy in the CCF probability of PMS component interface modules for the recirculation squib valve (V-118). The applicant immediately initiated corrective action. In RAI-SRP19.0-SPLA-07, the staff requested correction of the discrepancy and an updated report of PRA results. Specifics

requested included: (1) the results of re-solving and re-quantifying the baseline and RTNSS full-power PRA, the shutdown PRA, and the external events PRA; (2) new risk insights identified during requantification of the previously mentioned PRAs; and (3) the results of the revised importance analysis.

In a letter dated August 21, 2008, the applicant reported that, in addition to correcting the discrepancy, analysts identified measures to improve the realism of the PRA for I&C systems. Specifically, the analyses found the values selected for PMS and PLS component common-cause beta factors to be overly conservative. The applicant has revised the model to reflect component-specific common-cause beta factors for PMS and PLS system components modeled in the PRA.

In the same letter, the applicant committed to revising TR-102 to reflect: (1) the results of re-solving and re-quantifying the baseline and RTNSS full-power PRA, the shutdown PRA, and the external events PRA; (2) any new risk insights identified during requantification of the previously mentioned PRAs; and (3) the results of the revised importance analyses. The applicant also reported that "requantification of the at-power PRA indicate[s] that the core damage frequency (CDF) and large release frequency (LRF) values and top cutsets closely compare with these items documented in the [previously submitted Level 1 internal events] PRA...."

In a letter dated November 6, 2008, the applicant reported some results of model correction and requantification. (The software used for this process automatically re-solves the model each time the model is requantified.) The letter reported only those results of the PRA that were point estimates of CDF and LRF (total, at power, shutdown, and sensitivity to nonsafety SSCs). The applicant reported no other changes in risk insights or importance analysis. The staff noted several changes that met the criteria of DC/COL-ISG-3 during the October 2009 onsite audit of the PRA. For example, the applicant had not reported a shutdown PRA sequence of significance. In another case, design improvements had eliminated a risk-significant component. Because these and similar changes were not reflected, the staff did not find this letter to be fully responsive. The staff identified the absence of some corrected results in the DCD as Open Item OI-SRP19.0-SPLA-07.

In a letter dated December 17, 2009, the applicant proposed revisions to the DCD. The staff's evaluation of these proposed revisions and resolution of the open item is documented in Section 19.1.4.1, "Level 1 Shutdown Internal Events Probabilistic Risk Assessment."

19.1.2.1.11 Large Pressurizer and Low-Power Density

The AP1000 pressurizer is large in comparison to the pressurizer of currently operating plants. This reduces the frequency of reactor scrams by increasing transient operation margins. This feature also moderates the pressure rise during certain transient events, such as loss of main feedwater, thus reducing the likelihood of a challenge to the primary safety valves. A larger pressurizer volume, as compared to currently operating plants, also helps lower the peak pressure that can be reached after a postulated ATWS event.

The applicant found it necessary to alter the design of the pressurizer. TR-36 details this change. Section 5.4.5 of this report documents the staff's evaluation of this design change. The applicant did not propose a change to the PRA because of this modification.

Because the applicant analyzed the proposed design changes to the pressurizer and found that they do not alter system-level thermal-hydraulic response or success criteria, the staff concludes that no change to the internal events PRA model is necessary. However, this change had an impact on the seismic margin analysis. The staff's evaluation of this issue is addressed in Section 19.1.5.1, "Probabilistic-Risk-Assessment–Based Seismic Margin Analysis."

19.1.2.2 Special Advanced Design Features for Core Damage Consequence Mitigation

The following design features improve the ability of the containment to accommodate the challenges associated with severe core damage accidents. The AP1000 PRA and supporting deterministic analyses model the impact of these features on severe accident mitigation and containment performance.

19.1.2.2.4 External Reactor Vessel Cooling

To accommodate the higher decay heat level in the AP1000, Westinghouse needed to refine the AP600 reactor vessel insulation system (RVIS). The design was modified to increase the critical heat flux (CHF) at the surface of the reactor pressure vessel (RPV), enhancing the heat transfer through the RPV to the surrounding water. TR-24, APP-GW-GLR-060, "Reactor Vessel Insulation System - Verification of In-Vessel Retention Design Bases," February 2007, addresses COL Information Item 5.3-5 by verifying that reactor vessel insulation is consistent with the design bases established for in-vessel retention of a damaged core. COL Information Item 5.3-4 requires a structural analysis of the AP1000 reactor vessel insulation and support structure. TR-24 reports relevant results of that analysis.

The effectiveness of external reactor vessel cooling in the AP1000 design depends, in part, on a RVIS that provides an engineered pathway for supplying water cooling to the vessel exterior and venting steam from the reactor cavity during severe accidents. It is designed to limit thermal losses during normal operations. Section 5.3 of this report documents this design, which is discussed in Section 19.1.8.24 and evaluated in Section 19.2.3.3.1.3.2.

In RAI-TR24-SPLA-06, the staff noted that some paints and coatings used to protect the reactor vessel during shipping could have detrimental effects on CHF performance. In TR-106, the applicant stated that the external surface of the reactor vessel is bare metal. AP1000 DCD Section 5.3.4.5 now reflects the fact that a temporary protective coating applied before shipment will protect carbon steel surfaces. In DCD Section 19.34.2.1, the applicant stated that the vessel will have no coatings on the outside surface of the reactor vessel. This ensures that wetability of the surface will not be inhibited and CHF performance will not be degraded; the staff finds this acceptable. The COL licensee must remove these temporary coatings. The staff requested additional basis for confidence that this will be accomplished.

In a letter dated April 14, 2009, the applicant clarified the nature of the protective covering, which is to be an industrial form of shrink wrap that will be removed in the receiving process. The staff agrees that no additional controls are required; the statement in DCD Section 19.34.2.1 is sufficient and RAI-TR24-SPLA-06 is resolved.

19.1.2.3 Residual Risk from Changes Not Explicitly Modeled

The applicant reviewed all design changes for their potential to affect risk. TR-135 documented the process used for this review as well as the results of that process. The staff noted that, if a

design change proposal (DCP) dealt with an SSC modeled in the PRA, the applicant evaluated its potential to affect the PRA results. However, the applicant did not necessarily evaluate other changes that may have an impact (e.g., changes to assumptions or PRA insights, as well as changes to model logic or changes that may alter probabilistic parameter estimates).

For example, a new or revised operating procedure might alter, for some modes, the alignment of an SSC in a manner that is inconsistent with documented insights or assumptions. The equipment would then require realignment to prevent or mitigate the consequences of an event applicable to the mode in question. It may be appropriate to model a different basic event (and supporting SSCs) in the PRA model for that mode. Alternatively, additional constraints or conditions to control risk may be appropriate before initiating the proposed procedure. The staff expects the applicant to perform such assessments, even if it will usually result in a determination that no explicit model change or procedural constraint is necessary.

In RAI-SRP19.0-SPLA-05, the staff identified the specific example of vacuum fill operation and asked the following:

> Please identify and briefly describe each DCP incorporated in the amended design and assess its potential to have such an impact on the PRA. For each DCP that may have an impact upon the PRA or other risk studies (e.g., seismic and internal fire) please evaluate and report its potential significance.

In a letter dated September 5, 2008, the applicant provided the results of its evaluation of the vacuum fill operation. In addition, the applicant reported that it made a change in the process used for future DCPs. The applicant reviewed each one for its impact on the PRA, with no initial screening for PRA-modeled SSCs. This effort included a documented review of the PRA assumptions and PRA insights affected, potential changes to model logic and probability data, the effect of operational changes on component modeling, and the impact on other risk studies (e.g., seismic and internal fire). The documentation of the review identifies and briefly describes every DCP and provides the rationale used to determine its impact on the plant risk and changes to the PRA or other risk studies (e.g., seismic and internal fire).

The staff concludes that the applicant performed an appropriate evaluation of the risk implications of vacuum fill operations and that the design change process will provide adequate assurance that the risk implications of all changes after Revision 17 of the DCD will be assessed. In a letter dated April 23, 2009, the applicant provided a schedule for re-evaluation (using the revised criteria) of all DCPs processed to date: changes reflected in Revision 17 of the DCD will be re-evaluated first and re-evaluation of all earlier changes will be documented prior to initial fuel loading. The results will be reflected in the plant-specific PRA as upgraded and updated prior to initial fuel load. The COL licensee is responsible for these updates as part of COL Information Item 19.59.10-2. The staff finds that this provides adequate assurance that the risk implications of all changes will be appropriately assessed and, if necessary, analyzed. This is an acceptable method for controlling residual risk from changes that are not explicitly modeled in the internal events PRA. The staff considers RAI-SRP19.0-SPLA-05 resolved.

19.1.3 Safety Insights from the Internal Events Risk Analysis (Operation at Power)

Safety insights from the internal events Level 1 PRA include the following:

- dominant accident sequences contributing to CDF

- areas in which certain AP1000 design passive and defense-in-depth features were the most effective in reducing risk as compared to currently operating reactor designs
- major contributors to the estimated CDF from internal events, such as hardware failures, system unavailabilities, and human errors
- major contributors to maintaining the built-in plant safety (to ensure that risk does not increase unacceptably)
- major contributors to the uncertainty associated with the estimated CDF
- sensitivity of the estimated CDF from internal events to: (1) potential biases in numerical values; (2) assumptions made; (3) lack of modeling details in certain areas; and (4) previously raised safety issues

Safety insights from the internal events Level 2 PRA include the following:

- core damage sequences and accident classes contributing to containment failure
- frequency and conditional probability of containment failure
- leading contributors to containment failure and risk

19.1.3.1 Level 1 Internal Events Probabilistic Risk Assessment

In TR-102, the applicant described the results of the PRA that it had upgraded and updated to conform to the amended design.

The staff conducted an audit of the PRA model upgrade and update. The staff reviewed the qualification of the PRA staff, procedures used for conversion, and processes for updating the PRA. In addition, the staff examined the PRA model itself with emphasis on new fault trees developed for I&C systems. Chapter 7 of this report documents the staff's review of I&C system design changes. The staff also reviewed the electrical system model changes for consistency with Revision 1 of TR-79. Chapter 8 of this report documents the staff's review of electrical system design changes.

The staff found that the development of the I&C model was consistent with the amended I&C design, as described in TR-39, APP-GW-GLN-004, "Instrumentation and Control Design Change," May 2006; WCAP-16675-NP, APP-GW-GLR-071, "AP1000 Protection and Safety Monitoring System Architecture Technical Report," Revision 2, May 2009; APP-GW-GLR-018, "Failure Modes and Effects Analysis and Software Hazards Analysis for AP1000 Protection System," June 2006; and TR-97, Revision 1. TR-80, APP-GW-GLR-080, "Mark-up of AP1000 Design Control Document Chapter 7," October 2007, documents the impact of these design changes on the DCD. Chapter 7 of this report documents the staff's review of the I&C design changes. (Many of the design changes had no impact on the PRA and, therefore, no impact on the severe accident analysis, as documented in TR-102 and TR-135.)

In a letter dated November 6, 2008, the applicant reported some results of the PRA model requantification. The applicant estimated the mean CDF for the AP1000 design from internal events during operation at power to be about 2.41×10^{-7} per year, unchanged from what the

previous PRA reported. The applicant characterized this as equivalent, given appropriate treatment of uncertainties.

Although the applicant did not modify the initiating event frequencies in the model, the contribution of each initiating event to CDF changed slightly. The applicant reported that these changes were associated with the I&C model revision and updated electrical power dependencies. The applicant's assessment suggested that the changes were not of sufficient magnitude to alter the risk insights derived from the PRA results. For example, the top ten cutsets were identical, and the CDF attributable to failure of the most risk-significant system (the PMS) changed by only a small factor (i.e., it became about half as significant). Various LOCA initiating events continue to dominate the CDF profile (about 85 percent), followed by reactor vessel rupture (about 4 percent) and transient events (about 4 percent). Contributions from steam generator tube rupture (SGTR) events are slightly higher (about 4 percent), while ATWS sequences and loss of offsite power/station blackout events contribute even less than before (less than 1 percent).

Based on these results and the audit that provided confidence in the model upgrade and update process, the staff finds that the amended Level 1 internal events PRA at power did not change significantly. The staff finds that a plant-specific PRA report that is identical to the PRA for the certified design continues to provide an acceptable basis for risk insights and assumptions related to internal events.

However, changes to the design have altered some of the insights derived from the PRA (e.g., improving the design by eliminating a risk-significant SSC). As discussed in Section 19.1.2.1.10, in RAI-SRP19.0-SPLA-07, the staff requested the results of re-solving and requantifying the baseline and RTNSS full-power PRA, shutdown PRA, and external events PRA, as well as the results of the revised importance analyses. In a letter dated August 21, 2008, the applicant committed to re-solve and requantify the model after making some corrections. In a revised response dated November 6, 2008, the applicant altered this commitment, as discussed in Section 19.1.2.1.10, above, where it is identified as Open Item OI-SRP19.0-SPLA-07. Resolution of the open item is documented in Section 19.1.4.1 of this report.

19.1.3.2.3.3 Human Actions

In WCAP-16555, the applicant reviewed human actions with respect to risk achievement worth (RAW) and risk reduction worth (RRW). In addition, the applicant reviewed human actions required for maintenance, test, inspection, and surveillance (MTIS) support. The applicant stated that, on a deterministic basis, no human actions were required to mitigate any design-basis accident (DBA) or to prevent core damage following a DBA.

The applicant also identified 19 human actions as most significant from a probabilistic standpoint, though none of them came within an order of magnitude of the criteria previously accepted for a "critical" human action. The applicant added the following three human actions because an expert panel considered them to be significant:

1. Failure to recognize the need for and failure to isolate the RNS system, given rupture of the RNS piping when the plant is at hot/cold conditions (RHN-MAN04): The applicant added this action because of the short time available for the operator to act and the conflicting goals (maintaining core cooling by the RNS versus isolating a leak or break in the RNS piping).

2. Failure to recognize the need for and failure to actuate the hydrogen control system, given core damage following a LOCA (VLN-MAN01): The applicant added this action because its limiting RAW is relatively close to the criteria, and it is a function within the scope of RTNSS.

3. Failure to close equipment hatch and personnel airlocks following core damage during a shutdown event: The applicant added this action because human action importance could not be calculated for shutdown, internal events, or LRF. The expert panel considered that, under these conditions, the largest risk of large release would come from failure to close the containment. Closing the containment under these conditions involves closing the equipment hatch, personnel hatches, and temporary penetrations.

As noted by the staff, DCD Table 19.59-18 documents that it is important to maintain the ability to close containment hatches and penetrations during MODE 5 and MODE 6 before steam is released into the containment. There is a commitment for procedures and training to ensure that this action will be taken when required.

The staff found that the results were consistent with the methodology prescribed for the certified design and that the applicant conservatively identified risk-important human actions. For these reasons, the staff finds the results to be consistent with NUREG-0800 and, therefore, acceptable.

19.1.3.3.3 Important Insights from Level 3 PRA and Supporting Sensitivity Analyses

The applicant deleted the discussion of Level 3 PRA from Tier 2. The Level 3 PRA is now described only in the environmental assessment.

19.1.4 Safety Insights from the Internal Events Risk Analysis for Shutdown Operation

19.1.4.1 Level 1 Shutdown Internal Events Probabilistic Risk Assessment

The staff compared the results of the shutdown PRA, as seen in the current model, with the results reported in Revision 16 of the DCD (unchanged in Revision 17). Many of the results significantly differ from those reported in DCD Section 19.59.5.1, "Summary of Shutdown Level 1 Results." For example, Section 19.59.5.1 discusses the dominant sequences and key contributors to risk. The staff compared this documentation to the top 15 cutsets and the top 20 component basic events ranked by RAW from the CAFTA results. Loss of component cooling (supplied by the service water system [SWS]) or service water (supplied by the circulating water system [CWS]) during drained conditions contributes at least 73 percent to the CDF, as seen in the CAFTA results, versus 64 percent as reported in the DCD. Loss of the RNS initiating event during drained conditions contributes at least 10 percent to the CDF as compared to 6 percent reported in the DCD. Inadvertent draining through valve V024 (IEV-LOCA24ND) contributes more to the CDF than the risk of RCS overdraining, as seen in the CAFTA results. However, the DCD does not report this event. Some of these changes appear to meet the importance criteria of DC/COL-ISG-3 and, therefore, should be documented.

In RAI-SRP19.0-SPLA-13, the staff asked the applicant to update DCD Table 19.59-15, "Summary of AP1000 Results," and to provide the following information:

1. a list of cutsets for the AP1000 shutdown PRA that contribute to 95 percent of total shutdown CDF and any that contribute as much as 1 percent of total shutdown CDF

2. a list of all SSCs in the shutdown PRA with their RAWs (if RAW greater than 2)

3. a list of all human actions modeled in the shutdown PRA with their RAW

4. a list of all CCFs in the shutdown PRA with their RAW (if RAW greater than 2) (or confirmation that all are described in WCAP-16555)

In a letter dated August 21, 2008, the applicant stated that the next revision of TR-102 would include the PRA model changes discussed in response to RAI-SRP19.0-SPLA-13. In a subsequent letter dated November 6, 2008, the applicant stated that it would revise TR-102 but proposed no changes to the DCD. The NRC staff identified this as the first part of Open Item OI-SRP19.0-SPLA-13.

The staff conducted an audit of the corrected and amended PRA model at the applicant's offices on October 13–15, 2009, as documented in an audit report dated November 22, 2009. The staff determined that identified deficiencies in the PRA model had been corrected. In a letter dated December 17, 2009, the applicant proposed revisions to the DCD to amend the description of shutdown PRA results. The staff finds that the proposed changes to the DCD are consistent with DC/COL-ISG-3 and, therefore, acceptable. Therefore, Open Item OI-SRP19.0-SPLA-07 (addressed in Section 19.1.2.1.10) and Open Item OI-SRP19.0-SPLA-13 are considered resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19.1.4.2 Dominant Accident Sequences Leading to Core Damage

In RAI-SRP19.0-SPLA-13, the staff also asked the applicant to confirm that the list of major contributors to risk for each sequence that contributes more than 1 percent to the shutdown CDF remains consistent with the cutset results and to revise the DCD as necessary to describe all such sequences.

In the August 21, 2008 letter, the applicant stated that the next revision of TR-102 would include the PRA model changes discussed in this RAI-SRP19.0-SPLA-13 response. In a subsequent letter dated November 6, 2008, the applicant stated that it would revise TR-102 but proposed no changes to the DCD. The staff identified this as the second part of Open Item OI-SRP19.0-SPLA-13.

In a letter dated December 17, 2009, the applicant proposed revisions to the DCD. The staff's evaluation of these proposed revisions and resolution of Open Item OI-SRP19.0-SPLA-13 is documented in Section 19.1.4.1, above.

19.1.4.3 Risk-Important Design Features

The applicant now describes actuation of IRWST injection as the result of a fourth-stage automatic depressurization system (ADS) signal rather than a low hot-leg level signal. In the

first part of RAI-SRP19.0-SPLA-04, the staff asked the applicant to clarify the impact of this modification on the shutdown risk assessment.

The applicant responded in a letter dated July 22, 2008, that it had modified the logic description in the DCD to represent more clearly how the system is intended to function during shutdown conditions.

Appendix 19E to DCD Revision 15 described the logic in the AP1000 as follows:

- actuation of IRWST injection on low (empty) hot-leg level on a two-out-of-two basis (RCS hot-leg level channel basis)

- actuation of fourth-stage ADS valves on low (empty) hot-leg level on a two-out-of-two basis (RCS hot-leg level channel basis)

The DCD provides the following revised description:

- actuation of fourth-stage ADS valves on low (empty) hot-leg level on a two-out-of-two basis (RCS hot-leg level channel basis)

- actuation of fourth-stage ADS causes actuation of IRWST injection

This logic configuration forms the basis for the PRA model and shutdown risk assessment. The change in wording for the logic for actuation of IRWST injection has no impact on the results and insights of the shutdown risk assessment.

The staff agrees that the clarification did not alter the functional response of the system and confirmed that the change in description did not affect shutdown PRA insights. The staff considers the first part of RAI-SRP19.0-SPLA-04 resolved.

COL Information Item 18.7-1 includes the following statement:

> Since inadvertent opening of RNS valve V024 results in a draindown of RCS inventory to the IRWST and requires gravity injection from the IRWST, the COL applicant will have administrative controls to ensure that inadvertent opening of this valve is unlikely. The control room design will take into account this error.

In RAI-SRP19.0-SPLA-09, the staff requested information on the features of the control room design that will ensure that inadvertent opening of valve V024 is unlikely.

In a letter dated August 21, 2008, the applicant responded that, during shutdown, electrical power to valve V024 is blocked (breakers open and manually locked out) when RNS is in operation. This prevents inadvertent operation when the RCS would be depressurized. (ADS valves are open during shutdown conditions in accordance with Technical Specification 3.4.13.)

A permissive signal (valves V001A/B and V002A/B are fully closed and valve V023 is open) is required to permit manual opening of this valve V024. DCD Figure 7.2-1 shows a corresponding interlock to open the RNS hot-leg suction isolation valves, which is prevented if the IRWST cross-connects to the RNS (valves V023 and V024) are not fully closed.

The staff finds this to be an acceptable method of ensuring that inadvertent opening of a valve V024 is unlikely. The staff considers this portion of COL Information Item 18.7-1 to be closed and RAI-SRP19.0-SPLA-09 resolved.

19.1.4.3.2 Loss-of-Coolant Accidents during Safe Shutdown or Cold Shutdown or Both with the Reactor Coolant System Intact

The applicant modified the containment recirculation design to provide large, interconnected screens without separate trash racks or coarse and fine screens. Section 6.2.1.8 of this report documents the staff's assessment of this change. In the second part of RAI-SRP19.0-SPLA-04, the staff asked the applicant to clarify the impact of this modification on the shutdown risk assessment.

The applicant, in a response dated July 22, 2008, stated that the DCD reflects the use of large, interconnected recirculation screens for recirculation flow. The passive core cooling system (PXS) has two banks of interconnected screens that filter recirculation flow. The staff evaluated these screens using the guidance in RG 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," November 2003. The screens are constructed of perforated stainless steel plate that is used to form pockets. Actions taken to close Generic Safety Issue 191, "Experimental Studies of Loss-of-Coolant-Accident-Generated Debris Accumulation and Head Loss with Emphasis on the Effects of Calcium Silicate Insulation," May 2005, and to respond to generic letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004, resulted in more robust sump screen designs. As is the case with current operating plants, the structural integrity of the AP1000 screens precludes the need for trash racks. TR-147 also discusses the design of the AP1000 screens. Section 6.2.1.8 of this report documents the staff's evaluation of the screens.

The applicant judged the changes in the screen design to have no negative impact upon the PRA results; thus, the DCD PRA was not changed to reflect modifications to the screens. The applicant judged these changes to have a positive impact upon the PRA, with a lower failure probability of the screens resulting from the enhanced design and increased flow area. The DCD PRA did not credit this change in failure probability.

The staff confirmed that the PRA appropriately modeled the CCF of the screens, using a conservatively large beta factor. For the DC amendment, the modeling is conservative and, therefore, acceptable to the staff. The staff considers the second part of RAI-SRP19.0-SPLA-04 resolved.

19.1.4.3.6 Loss of the Normal Residual Heat Removal System (due to Loss-of-Coolant Accidents) or Loss of the Normal Residual Heat Removal System or Its Support Systems during Reactor Coolant System Open Conditions

When the RCS is open, the importance of the RNS is higher than at other times because safety-related heat removal paths may not be available. An external event (high winds) can have an impact on alternating current power sources required for RNS function because those sources (and their fuel) are not protected by safety-related structures. In addition, if the RNS becomes unavailable, the containment must be closed before boiling begins in the RCS.

In RAI-SRP19.0-SPLA-18, the staff asked the applicant to evaluate high winds while in MODE 5 and MODE 6 or to provide an acceptable basis for screening such events from consideration.

The associated risks should be quantified and possibly controlled. The NRC staff identified this as OI-SRP19.0-SPLA-18.

In a letter dated March 26, 2009, the applicant addressed high wind events occurring in MODE 5 and MODE 6. The applicant stated that emergency response requirements or emergency action levels will require that the RCS be taken out of mid-loop operation and prohibit entry when a potentially severe high wind event is anticipated. In addition, the response describes how core cooling is accomplished if diesel generators are not available.

The staff finds that the proposed measures are appropriate and sufficient to justify screening high wind events during MODE 5 and MODE 6 from further analysis. Controls on the implementation of emergency response requirements are an acceptable way to ensure that these measures will be implemented. The staff considers Open Item OI-SRP19.0-SPLA-18 to be closed.

19.1.5 Safety Insights from the External Events Risk Analysis

Three sections of the AP1000 DCD address PRA of external events consistent with DCD Section 1.9.5.2.14:

1. A risk-based seismic margin analysis (SMA), documented in DCD Section 19.55 and Appendix 19A, both titled "Seismic Margin Analysis," addresses seismic events. Sections 19.1.5.1 of this report document the staff's evaluation of the SMA.

2. APP-GW-GL-022, "AP1000 Probabilistic Risk Assessment," of July 2004, Revision 8, Chapter 57, "Fire Risk Assessment," documents analysis of the risk associated with internal fires. (The analysis is not discussed in this report because it has not changed since initial certification of the AP1000 design.)

3. DCD Section 19.58, "Winds, Floods, and Other External Events," addresses remaining external events. Sections 19.1.5.4 through 19.1.5.7 of this report document the staff's evaluation.

The objectives of the external events risk analysis provided in Section 19.58 of the AP1000 DCD are threefold:

1. Determine screening criteria and identify potential external events that may affect the AP1000 risk on a site-specific basis.

2. Provide generic risk analyses, based on bounding assumptions regarding site-specific parameters (e.g., frequency of each category of hurricanes) for relevant external events.

3. Provide guidance to COL applicants regarding the verification of the applicability of these generic analyses to a specific site.

The AP1000 DCD addresses those external initiating events or external hazards whose causes are external to the plant, other than seismic events. Based on the modified individual plant examinations of external events (IPEEE) guidelines, DCD Section 19.58 discusses the following external events or external hazards:

- high winds (including tornadoes)

- external floods
- external fires
- transportation and nearby facility accidents

The scope of this analysis does not include sabotage, which is consistent with the SRP and therefore acceptable to the staff. The information provided in DCD Section 19.58 is based primarily on the following:

- NRC guidance for the preparation and submittal of IPEEE for operating nuclear power plants

- the AP1000 DC PRA

- site-specific information related to external events for several proposed sites to build a nuclear plant referencing the AP1000 design

On June 28, 1991, the NRC issued Supplement 4 to GL 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," requesting that each licensee conduct an IPEEE. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events for Severe Accident Vulnerabilities," documents NRC guidelines for conducting IPEEE and on the structure and content of the IPEEE submittal. The staff examined these guidelines to verify their applicability to new reactor licensing and to investigate their completeness. The staff concludes that the IPEEE guidelines are applicable to the process of COL application after they are properly modified as follows:

- The IPEEE, performed by operating reactor licensees, takes into account plant-specific licensing information regarding external hazards that is not necessarily available to a COL applicant. For example, NUREG-1407 states, "[the] effects of external fires, other than loss of offsite power (LOSP), have been evaluated during the operating license (OL) review against sufficiently conservative criteria." Thus, the assumptions made in NUREG-1407 (e.g., in deriving the list of external events to be included in the IPEEE submittal) must be examined to determine whether a COL applicant must address events that were not included in the DCD.

- The baseline risks of the AP1000 design, as assessed in the DC PRA, are lower than the corresponding risks of an average operating plant. At operating reactors, the combined CDF for external events may be of the same magnitude as CDF for internal events. For this reason, the criteria for screening external events from the quantitative evaluation must be properly adjusted to maintain the conclusion reached in the DC—that the AP1000 design represents a reduction in risk compared to existing plants.

The applicant gathered site-specific external events information from utilities interested in the AP1000 design and performed a generic analysis for each external event, based on the most limiting parameters from any site. In TR-101, the applicant identified potential external events that may affect the AP1000 risk.

The staff finds that these external hazards are most likely a complete list of external events associated with candidate sites for an AP1000 plant as of the date of this report. The staff evaluated the analysis of the AP1000 response to external events according to the SRP, which states that the applicant's analyses should be "comprehensive in scope and address all

applicable...external events and all plant operating modes." The staff requested additional information in RAI-TR101-SPLA-01 through RAI-TR101-SPLA-08 to clarify the report. In a letter dated October 19, 2007, the applicant responded. The staff requested additional clarification on RAI-TR101-SPLA-03 (external fires) and RAI-TR101-SPLA-06 (external flooding), which was provided in a letter dated February 8, 2008.

The applicant added consideration of external fires to the external events PRA and provided additional information on flooding caused by storm surge.

The methods used by the applicant to analyze external hazards, as documented in TR-101 and described in the DCD, are consistent with RG 1.200. Therefore, the external events analysis is acceptable to the staff given the input parameters used. There are two exceptions: the release of hazardous materials from nearby facilities and the treatment of high winds.

The analysis neither included an explicit discussion of the release of hazardous materials from nearby facilities (other than pipelines) nor identified this issue as a COL information item. The staff is concerned that some toxic materials are immediately dangerous to life and health at concentrations lower than the materials evaluated for pipelines, and some may not be readily detected. In RAI-SRP19.0-SPLA-17, the staff requested an assessment of risk from the release of toxic materials and a basis for a COL applicant to confirm that the assessment bounds the risk at the proposed site.

In a letter dated March 9, 2009, the applicant addressed the release of hazardous materials from nearby facilities and provided justification for screening of toxic releases from further analysis. The applicant stated that no operator action was credited, obviating the need to evaluate specific toxic release events with respect to type and amount of material released. The result of the analysis was a conditional core damage probability of 6.26×10^{-8}. From this, a limiting event frequency was provided for COL applicants to use to confirm that the generic analysis is applicable to their proposed sites. The applicant identified several conservatisms in this analysis. In addition to the assumption that operators were immediately and completely unable to perform any protective or mitigating actions, design features that assure control room habitability for 72 hours, under nearly all circumstances, were not credited.

In the same response, the applicant clarified the basis for using an initiating event frequency of 1×10^{-6} for the analysis of marine explosions. The applicant confirmed that screening of the event was based on a negligible contribution to core damage frequency so long as the criteria of RG 1.91, "Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants," Revision 1 are met.

The staff agrees that there is considerable conservatism in the analysis of toxic gas release events that was described, and finds that it provides an acceptable basis for screening such events from further risk assessment. The limiting event frequency for toxic releases in the DCD provides an appropriate basis for COL applicants to confirm that this analysis bounds conditions where they propose to build a plant that references the AP1000 certified design. The staff considers the description of external events from transportation and nearby facility accidents to be complete and RAI-SRP19.0-SPLA-17 is resolved.

The applicant did not address the case of high winds while in MODE 5 and MODE 6. This scenario should be screened from consideration or the associated risks quantified and possibly controlled. The applicant addressed this concern in the March 26, 2009 response to RAI-SRP19.0-SPLA-18. The staff's evaluation is provided in Section 19.1.4.3.6 of this report.

The COL applicant must verify that the generic analysis for each external event bounds conditions at the proposed site. In a letter dated August 23, 2010, the applicant provided a COL information item to ensure that this action is taken by COL applicants.

DCD Section 2.2 requires a COL applicant to identify design changes in its safety analysis report if the occurrence of an external event that leads to severe consequences is 1×10^{-6} per year or greater. Accordingly, the COL applicant should assess the risk associated with any safety hazard that does not meet this criterion for being screened from further evaluation. In addition, the licensee must reevaluate the external event risk when a site-specific, plant-specific PRA is available.

The criteria for screening out external events from the quantitative evaluation are adjusted to maintain (for a plant referencing the AP1000 design) the conclusion reached in the DC—that the AP1000 design represents a reduction in risk compared to existing plants. The AP1000 DCD uses the following criteria with respect to risk evaluation of external events or hazards:

- An event or hazard with frequency less than 1×10^{-7} per year is screened from further evaluation.

- An event or hazard with frequency of 1×10^{-7} per year or higher is screened from further evaluation if a qualitative or bounding analysis shows that the associated CDF is less than 1×10^{-8} per year.

- An event or hazard with frequency of 1×10^{-7} per year or higher that cannot be shown to contribute less than 1×10^{-8} per year to CDF must be addressed in the risk analysis.

Each COL applicant must confirm that the high winds, floods, and other external events analysis documented in the DCD are applicable to the site for which the COL application is submitted (i.e., the spectrum of events at the site is bounded by the events analyzed in the DCD). Chapter 19 of the COL final safety analysis report (FSAR) should document this applicability evaluation. Further evaluation will be required if any unbounded, site-specific susceptibilities are found.

The NRC requires, where applicable to the site, that the COL applicant perform a site-specific, PRA-based analysis of external flooding, hurricanes, or other external events pertinent to the site to reveal any site-specific vulnerabilities. It is sufficient for the COL applicant to provide the basis for a conclusion that, for the proposed site, a particular external event is no more frequent and no more severe than that same event as modeled for the certified design. The COL licensee must develop plant-specific and site-specific risk information before loading fuel. This is part of COL Information Item 19.59.10-2.

In addition, the PRA used to support the AP1000 DC will be updated, as necessary, when site-specific and plant-specific (as-built) data become available. The COL applicant will review differences between the as-built plant and the design used as the basis for the AP1000 PRA to determine whether the PRA results are significantly impacted. Special emphasis should be placed on areas of the design that either were not part of the certified design or were not detailed in the certification. This is part of COL Information Item 19.59.10-2.

In RAI-SRP19.0-SPLA-02, the staff asked the applicant to clarify how SSCs are designed to withstand the effects of flooding. In its July 22, 2008, response letter, the applicant responded

that the AP1000 is protected against floods up to the 30.5 meter (m) (100-foot (ft)) level. The 30.5 m (100-ft) level corresponds to the plant ground level. From this point, the ground is graded so that water will naturally flow away from the structures. Additionally, all seismic Category I SSCs below grade (below ground level) are designed to withstand hydrostatic pressures, and they are protected against flooding by a water barrier consisting of waterstops and a waterproofing system.

The staff finds that the design of safety-related SSCs below the 30.5 m (100-ft) level provides adequate protection from the effects of external flooding. Section 3.4 of this report discusses the staff's evaluation of internal flooding.

The COL applicant referencing the AP1000 certified design is responsible for: (1) confirming in the COL application that the information provided in Section 19.58 of the DCD is applicable to the selected site; and (2) addressing all site-specific action items discussed in Section 19.58 of the DCD.

The staff concluded that the methods used in the AP1000 PRA to evaluate external events provide the insights necessary to determine whether any design or procedural vulnerabilities exist for these external events. These methods provide insights needed for DC requirements, such as ITAAC. The staff finds that, for the events specified in the DCD, the reported results are acceptable. However, the case of high winds while in a shutdown mode had not been addressed. The staff identified this as Open Item OI-SRP19.0-SPLA-18. Subsequently, additional information was provided by the applicant; the staff's evaluation is documented in Section 19.1.4.3.2 of this report.

19.1.5.1 Probabilistic-Risk-Assessment–Based Seismic Margin Analysis

All seismic Category I SSCs are designed to remain functional when subjected to an earthquake, defined in 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants" as the safe shutdown earthquake (SSE). 10 CFR 100.23, "Geologic and seismic siting criteria," also applies to site-specific seismic hazard. The seismic analysis and design of the AP1000 plant is based on the certified seismic design response spectra (CSDRS) shown in DCD Tier 1, Figures 1.0-1 and 1.0-2. The CSDRS are based on RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," with an enhanced spectral acceleration in the 25-hertz (Hz) region. Its peak ground acceleration (PGA) is 0.3 g, while its dominant energy content is in the frequency range of 2 to 10 Hz.

In the DC amendment, the applicant presented another set of seismic response spectra in DCD Tier 1, Figures 1.0-3 and 1.0-4. The results of the applicant's analyses for the new set of seismic response spectra are shown in TR-115, APP-GW-GLR-115, "Effect of High-Frequency Seismic Content on SSCs," Revision 1, October 2008. This is a seismic analysis of the AP1000 nuclear island using hard-rock, high-frequency (HRHF) spectra that bound three different site conditions in the central and eastern United States. In addition to the results of these linear-elastic analyses for the design basis load determination, the applicant conducted detailed foundation structure interaction analyses using an approved coherency function to account for the scattering effect of seismic input. The effect of incoherency in seismic input at different points on a large foundation slab tends to reduce response for SSCs with natural frequencies from 25 to 50 Hz. TR-115 also provides supplemental criteria for selection and testing of equipment whose function might be sensitive to high-frequency acceleration. Section 3.7 of this report discusses the staff's evaluation of the seismic design.

In SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," Section II.N, "Site Specific Probabilistic Risk Assessments and Analysis of External Events," the staff recommended a PRA-based seismic margin approach. At the plant level, high confidence in low probability of failure (HCLPF) should be established by ensuring that the seismic capacity of SSCs needed for safe shutdown is much larger than required for the design-basis earthquake. In a staff requirements memorandum dated July 21, 1993, the Commission modified that recommendation:

> PRA insights will be used to support a margins type assessment of seismic events. A PRA based seismic margins analysis will consider sequence level HCLPF [value]s and fragilities for all sequences leading to core damage or containment failures up to approximately one and two thirds the ground motion acceleration of the design-basis safe shutdown earthquake (SSE).

The applicant established a review-level earthquake (RLE) to demonstrate a one and two-thirds margin over the SSE, corresponding to the seismic design response spectra specified in DCD Tier 1, Figures 1.0-1, 1.0-2, 1.0-3 and 1.0-4.

For specific sites, the ground motion response spectra (GMRS) are obtained from site-specific, probabilistic seismic hazard–based analysis. Many of the GMRS of the central and eastern U.S. rock sites show response amplitudes that exceed the CSDRS for some frequency ranges. For this reason, an HRHF spectrum has been developed that bounds three hard-rock sites where AP1000 plants are proposed. DCD Figures 3I.1-1 and 3I.1-2 compare the HRHF at foundation level against the AP1000 CSDRS for both the horizontal and vertical directions for 5-percent damping. The HRHF spectrum exceeds the CSDRS for a range of frequencies above about 15 Hz. At high frequencies of vibratory excitation, the relative displacement is small and produces insignificant increase in stress. As an example, at 25 Hz and a spectral acceleration of 1.0g, the relative displacement is 0.016 inch. This is too small to cause damage.

In TR-115, the applicant evaluated representative SSCs that have been identified as "potentially sensitive to high-frequency input" in locations where the GMRS demonstrated an exceedance (magnitude greater than the CSDRS) in the high-frequency region.

In TR-144, APP-GW-GLN-144, "AP1000 Design Control Document High Frequency Seismic Tier 1 Changes," Revision B, December 2007, the applicant stated that additional equipment dynamic qualification effort beyond the seismic design bases for operating nuclear power plants to address high-frequency response effects is not warranted. However, the applicant noted that the effect of high-frequency input on potentially sensitive active components requires additional consideration in accordance with interim staff guidance DC/COL-ISG-1.

In TR-144, the applicant concluded that, for structures, the HRHF loads would not govern the design. For the primary component supports and reactor coolant loop nozzles, seismic loads from the CSDRS bound those from the high-frequency input. Consequently, the staff considered these items to be acceptable seismic design for the HRHF input. For piping systems, the applicant concluded that the results of the HRHF seismic analysis are bounded by the stress results of the AP1000 CSDRS seismic analysis.

For safety-related electrical equipment, the applicant concluded that the qualification methodology (analytical evaluations and testing procedures) currently employed generally leads to a more conservative design than that resulting from the HRHF spectra. Supplemental

seismic testing of high-frequency-sensitive safety-related equipment or implementation of one of the high-frequency screening techniques, approved in DC/COL-ISG-1, may be required to demonstrate acceptability under HRHF seismic demand conditions. Based on the acceptability of the seismic design and supplemental criteria for potentially susceptible equipment, the applicant stated that the conclusions of the PRA-based SMA are unchanged. In a recent update to seismic PRA based systems analysis, the applicant conducted a PRA sensitivity study ignoring the functions of all safety related electrical equipment; this study shows that the plant-level HCLPF value remains unchanged. This is primarily because of the passive design where key active components are designed to perform their safety functions when nonsafety-related support systems fail. This point is discussed further below.

Section 3.7 of this report discusses the staff evaluation of the seismic design. Subsequent to the application, the staff issued Interim Staff Guidance, DC/COL-ISG-20. This clarifies the staff's expectations for information to be included with an application for DC and the application for a license. It also identifies actions that the COL licensee must take to verify that the plant was built as designed.

For structures, primary component supports, and reactor coolant loop nozzles, as well as piping systems, the staff concludes that the applicant has demonstrated that, for the range of frequencies relevant to these SSCs, seismic loads are enveloped by CSDRS. This provides a sufficient basis for the staff to conclude that the seismic margins for these SSCs are acceptable.

In addition to the new parameters for the AP1000 generic site, the application for amendment to the certified design proposed a number of design changes that may affect the seismic capacity of equipment required to bring the plant to a safe, stable condition and to maintain containment integrity. Some proposed changes added seismic Category I SSCs to the design; others were deleted. These changes have the potential to affect the AP1000 SMA. Because of the changes, the staff requested an updated description of the results and insights of the AP1000 SMA.

Moreover, for safety-related equipment that is potentially sensitive to high-frequency excitation, the staff could not conclude that the applicant demonstrated adequate seismic margin, given the higher amplitude of high-frequency components of the GMRS. Although safety-related equipment that exhibits natural frequencies within the HRHF exceedance range will be subject to supplemental high-frequency seismic evaluation to confirm an acceptable seismic design, the applicant must clarify the basis for confirming that seismic margin is adequate.

The SMA for the certified design identifies HCLPF at the sequence level using a minimum-maximum[6] approach. Each COL applicant referencing the design should describe relevant site features and provide the basis for concluding that an acceptable seismic margin is maintained using this method or an alternative that is adequately justified. The staff identified these issues as Open Item OI-SRP19.0-SPLA-12.

In a letter dated August 23, 2010, the applicant provided a proposed revision to Section 19.55 of the DCD, "Seismic Margin Analysis," reporting the results of an updated, PRA-based SMA that

[6] In the minimum-maximum or "min/max" approach, in a sequence where the failure of any individual SSC would cause core damage, the lowest individual SSC HCLPF value is used to characterize the entire sequence. If there is a sequence where the failure of multiple SSCs must occur to result in core damage, the highest HCLPF value for any of those SSCs is used.

reflects the current site parameters for the standard design. In Section 19.59.10.5, "Combined License Information," new and revised COL information items were proposed. The staff finds that the proposed changes to the DCD are consistent with DC/COL-ISG-3 and DC/COL-ISG-20; therefore, Open Item OI-SRP19.0-SPLA-12 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19.1.5.1.1 Dominant Accident Sequences for Seismic Events

The applicant's risk-informed SMA identified accident sequences for seismic events. For the SMA, the dominant sequences and associated cutsets are those that limit HCLPF values for the plant, irrespective of their likelihood. The margins approach does not support the determination that these are important contributors to seismic risk in a probabilistic sense, but allows identification of the plant features that are important to the plant level HCLPF value. The redundancy and diversity available in achieving that HCLPF value can also be confirmed.

The PRA-based SMA shows that the AP1000 design meets or exceeds the 0.5g HCLPF value. The applicant also performed a bounding analysis, using simplified and conservative assumptions, to identify paths by which the containment could be bypassed, fail to isolate, or fail. This analysis assumed that the reactor vessel fails because of failure of the fuel (HCLPF value 0.5g) and that the containment fails if the reactor vessel fails. Thus, the plant HCLPF for large release is the same as for core damage. This is an artifact of the applicant's conservative approach to SMA. For the AP1000, the conditional containment failure probability is nearly a full order of magnitude less than 1.0; LRF is much less than CDF. Because of design features to ensure in-vessel retention of the core, these are very conservative assumptions. Nevertheless, the AP1000 satisfies the expectation of the Commission as expressed in SECY-93-087 that the plant HCLPF value will be at least one and two-thirds times the SSE. Therefore, the staff concludes that seismic risk for the AP1000 design is acceptable.

COL applicants will confirm that site-specific features do not reduce the HCLPF value of a proposed plant. The capacity of as-built SSCs will be confirmed by a seismic walkdown to be performed after construction; the plant HCLPF value must remain at least as high as the value for the certified design.

The updated SMA was performed in a manner that was essentially identical to the SMA for the certified design. The capacity of those components required to bring the plant to a safe, stable condition was assessed. In some cases, for a particular SSC, a different method of calculating the seismic capacity of an SSC was selected from among several options that are acceptable to the staff.

In the risk-based SMA, no credit is taken for nonsafety-related systems. Because such systems are not seismic Category I, it is assumed that they become unavailable as a consequence of the seismic initiating event. The HCLPF value associated with transmission line ceramic insulators is low (0.09g), so all seismic events are assumed to entail loss of off-site power. Since the diesel generators are nonsafety-related and, therefore, assumed not to be available, all seismic accident sequences involve station blackout (loss of all alternating current (ac) power). The analysis investigated and accounted for the potential for adverse interactions between nonsafety-related SSCs (assumed to be damaged) and safety-related systems. The event and fault trees developed for the internal events PRA were modified to accommodate seismic events. In this way, the random failures and human errors modeled in the internal events portion of the PRA are captured in the seismic analysis.

The modified event and fault trees were merged and cutsets for all sequences that lead to core damage were generated. Most of the HCLPF values for components and structures were obtained through computing a conservative deterministic failure margin (CDFM), performing a probabilistic fragility analysis, or using deterministic methods. The HCLPF values for electrical equipment, where test results are documented, were obtained by comparing required response spectra to test response spectra for similar types of equipment.

19.1.5.1.2 Risk-important Features and Operator Actions for Seismic Events

The limiting SSCs for which seismically induced failure would lead directly to core damage include the pressurizer and the fuel in the reactor vessel (a HCLPF value of 0.5 g). This HCLPF value was also assumed for a large number of SSCs that may be sensitive to high frequency excitation. By design, even if these frequency-sensitive components are determined to have a lower seismic capacity, the plant HCLPF value would remain the same (0.5 g).

Updated HCLPF values for most structures indicate higher seismic capacity. An exception is the polar crane: due to a change in the design loading of the crane, it has become the limiting SSC for gross structural failure (0.55 g). Although parts of nonsafety-related structures have been upgraded to seismic Category II, the SMA did not take credit for these changes. The staff finds this to be conservative and therefore acceptable.

Operator actions are not credited in the SMA model for events that control the plant HCLPF value. The staff concludes that this is conservative, as the inclusion of operator actions in the models would only have the potential to increase the plant HCLPF.

19.1.5.1.3 Insights from Uncertainty, Importance, and Sensitivity Analyses for Seismic Events

Uncertainty analysis for seismic events is performed to quantify the range of values within which the results of an analysis could reasonably be expected to fall. Hazard curves, the result of a seismic hazards analysis, have a large uncertainty due to variability in source term and attenuation relationships. The large uncertainties in the hazard curves will dominate the resulting core damage frequency analysis, but uncertainties in equipment and structure fragilities are subject to much lower variability, since more and better information exist for them. This makes sensitivity analysis for SSCs less meaningful. Consequently, staff finds that performing a convolution of seismic hazard and fragility is not necessary for a generic plant design; consequently, a sensitivity analysis of plant HCLPF to SSC fragility is not meaningful. No additional uncertainty analysis was performed because uncertainty is directly addressed in the margins method. HCLPF values represent the seismic capacity (peak acceleration expressed in terms of the acceleration of gravity) at which there is 95 percent confidence that equipment needed for safe shutdown will fail less than 5 percent of the time. This provides margin to bound uncertainty while avoiding the extremes of the probability distributions.

Because the margins method does not quantify risk, importance analyses were not performed. The applicant did, however, perform an additional sensitivity analysis on the effects of changes in certain assumptions used in the SMA. The applicant chose to vary the HCLPF values for equipment that is to be seismically qualified by testing. When these values are varied from 0.5 g to 0.3 g, the HCLPF value is not changed for any sequence or event. Consequently, the plant HCLPF value was also unchanged.

The AP1000 SMA has shown that the plant HCLPF value is at least one and two-thirds the ground motion acceleration of the design-basis SSE. Because it limits the plant HCLPF value (0.50 g), the nuclear fuel is included within the scope of the reliability assurance program (RAP). The pressurizer, although its upper support weld was assigned the same HCLPF value in the SMA, was not included in the RAP because an alternative method of analysis shows it to have a significantly higher HCLPF value (0.56 g, based on the upper support strut). Because the SMA takes no credit for nonsafety-related systems to mitigate seismic events, the results of the SMA do not affect the probabilistic criteria used to select nonsafety-related SSCs for inclusion in the RAP. The staff concludes that the result of PRA-based SMA, as amended to include hard-rock and soil sites, meets the Commission expectation for adequate margin expressed in its staff requirements memorandum (SRM) to SECY-93-087.

19.1.5.4 High Winds Evaluation

High winds can affect plant structures in two ways: (1) structures can collapse or overturn from the excessive loading when wind forces exceed the load capacity of the structure; and (2) lifting and thrusting can cause materials to act as missiles against plant structures that house safety-related equipment. In addition, the applicant investigated the potential for debris generated by high winds to clog the passive containment cooling system (PCS) drains and directly or indirectly to block the PCS air baffle.

The AP1000 structures protecting safety-related features are designed to withstand winds of up to 483 kilometers per hour (km/h) per hour (300 miles per hour (mph)), as well as missiles generated by these winds (see design-basis wind speed discussed in Chapter 2 of the DCD). Also, the AP1000 operating basis wind speed is 233 km/h (145 mph), as discussed in Chapter 2 of the DCD. In general, there is some margin above the design and operating bases that the risk evaluation of high winds neither assesses nor credits. The applicant made the following assumptions in evaluating the risk from high winds:

- Safety-related structures, which house safety-related equipment, are not impacted by high winds of any kind (tornados and hurricanes, including extra-tropical cyclones) if the wind speed does not exceed 483 km/h (300 mph).

- Nonsafety-related structures, which are designed and built according to uniform building code and house nonsafety-related defense-in-depth or investment protection equipment, are not impacted by high winds of any kind (tornados and hurricanes, including extra-tropical cyclones) if the wind speed does not exceed 233 km/h (145 mph).

- High-wind events exceeding 483 km/h (300 mph) are extremely rare events with a frequency of less than 1×10^{-7} per year; therefore, they are screened out from the risk analysis based on the screening criteria discussed in Section 19.1.5, above. The COL applicant referencing the AP1000 design must verify this assumption.

The applicant states in DCD Section 19.58.2.1 that no tornados or hurricanes are expected to reach 483 km/h (300 mph) winds per the enhanced Fujita scale for tornados and the Saffir-Simpson scale for hurricanes. Though the staff does not assign an upper wind speed limit to these scales, the conclusion is consistent with the staff's position documented in RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1. For the continental United States, the staff considers the highest tornado wind speed with a frequency of 1×10^{-7} to be 370 km/h (230 mph). AP1000 safety-related structures are designed to withstand winds of 483 km/h (300 mph). Clearly, the expected frequency of 483 km/h

(300 mph) tornadoes is significantly lower. For plants that are to be sited in the continental U.S., such events may be screened from further analysis.

Based on these assumptions, the applicant performed generic risk evaluations using initiating event frequencies that it described as "bounding." Six tornado event categories (defined in Table 19.58-1 of the DCD, which describes the enhanced Fujita scale for tornados) and five hurricane event categories (defined in Table 19.58-2 of the DCD, which describes the Saffir-Simpson scale for hurricanes) were evaluated. In addition, the applicant considered extra-tropical cyclones as a single category of high winds. Extra-tropical cyclones are normal storms and thunderstorms with winds expected to fall below the operating basis of 233 km/h (145 mph). The analysis assumed a bounding frequency of extra-tropical cyclones equal to 3×10^{-2} per year. COL applicants referencing the AP1000 design must verify that the frequency of each of the 12 high wind categories at the proposed site is bounded by the frequency assumed in Section 19.58 of the AP1000 DCD.

High winds cannot be screened out from the evaluation using the initiating event frequency criterion because the assumed frequency of high winds is greater than 1×10^{-7} per year. However, bounding risk assessments have shown that the CDF associated with high winds is less than the criterion of 1×10^{-8} per year. Therefore, the applicant did not perform detailed risk assessments for high winds. The applicant did perform risk assessments for three cases—a baseline case and two sensitivity cases:

- The baseline case assumes two kinds of failures: (1) an unrecoverable LOSP event for all 12 high wind categories since the site switchyard is unprotected; and (2) failure of all nonsafety-related structures for three categories of tornados (designated as EF3, EF4, and EF5 in Table 19.58-1 of the DCD) and three categories of hurricanes (designated as Category 3, 4, and 5 in Table 19.58-2 of the DCD), which exceed the operating basis of 233-km/h (145-mph) winds that could impact nonsafety-related structures. The applicant assumed that the failure of the nonsafety-related structures leads to the failure of all nonsafety-related systems credited in the AP1000 PRA with the exception of the manual DAS. The DAS manual actuation cables are located within the nuclear island and are, therefore, protected against high winds.

- The first sensitivity case assumes an unrecoverable LOSP event for all 12 high wind categories but no other failures. This sensitivity case removes the conservative assumption that all nonsafety-related structures fail when the operating basis of 233 km/h (145 mph) for high winds is exceeded, even though all structures are designed with some margin to withstand winds above the operating basis.

- The second sensitivity case assumes that all 12 high wind categories cause both an unrecoverable LOSP event and the failure of all nonsafety-related structures. This sensitivity case is a very conservative upper case since it assumes that all nonsafety-related structures fail even for categories of high winds that do not exceed the operating basis of 233 km/h (145 mph).

Table 19.58-3 of the AP1000 DCD summarizes the three risk assessment cases. The estimated CDF for the baseline case is about 5×10^{-9} per year, which is less than the criterion of 1×10^{-8} per year. Therefore, no detailed risk assessment of high winds is necessary. The CDF for the first and second sensitivity cases are about 2.3×10^{-9} and 1.4×10^{-8}, respectively. The first sensitivity case indicates that the estimated risk is not significantly sensitive to assumptions about the impact on nonsafety-related structures of high winds exceeding the operating basis.

The second sensitivity case indicates that the screening criterion of CDF less than 1×10^{-8} per year is almost met even under very conservative assumptions about the failure of nonsafety-related structures. The staff finds that the bounding risk assessments documented in Section 19.58 of the AP1000 DCD show that, under the stated assumptions for external events (which must be verified by the COL applicant), the risk from high winds (tornados and hurricanes) is so small that no detailed risk assessments for high winds are needed. In RAI-SRP19.0-SPLA-03, the staff requested the addition of COL information items to enumerate the assumptions to be verified. In a letter dated August 21, 2008, the applicant proposed to amend the DCD to state, "A site specific review of the generic PRA should be conducted to verify that the assumptions in the PRA bound the site specific conditions for the applicant's site." In a letter dated August 23, 2010, the applicant proposed a revision to Section 19.59.10.5, "Combined License Information," with a revised COL information item to make this explicit. The staff finds that the proposed change is consistent with RG 1.200 and acceptable; therefore, RAI-SRP19.0-SPLA-03 is resolved.

In addition to structural failures, the applicant investigated the potential for blockage or plugging of the PCS airflow path by debris generated by high winds. The applicant found that blockage or plugging of the PCS airflow path could occur by blockage of the screens, failure of the louvers, or blockage of the chimney outlet. However, because of the presence of certain design features and operational requirements, these failure mechanisms are highly unlikely (i.e., their frequency is smaller than 1×10^{-7} per year). For this reason, the PRA does not model them:

- Large openings are distributed around the circumference of the shield building, just underneath the roof. Screens and fixed, always-open louvers designed to prevent foreign objects or debris from entering the air flowpath cover these openings, providing an area through which air can flow into an enclosed plenum.

- Ducts made of pipe slant upward from this plenum through the shield wall. They supply air to a common plenum above the outer flow annulus inside the shield building.

- A walkway provides access for inspection of the flowpaths and removal of debris.

- There is a surveillance requirement to verify that the airflow path is unobstructed.

- The chimney outlet is designed to produce the necessary airflow in the event of an accident. The outlet contains heavy grates to guard against missiles. Screens prevent the entry of foreign objects into the annulus around containment. During normal operation, a positive airflow prevents ice and snow from entering the chimney.

In RAI-SRP19.0-SPLA-01, the staff requested that insights related to high winds and containment cooling be added to the DCD (RAI-SRP19.0-SPLA-01 also discussed external flooding as addressed in Section 19.1.5.5). In a letter dated July 22, 2008, the applicant proposed a revision to DCD Table 19.59-18, identifying these features and requirements among the PRA assumptions and insights. This is consistent with NUREG-0800 and is, therefore, acceptable to the staff.

In a letter dated March 4, 2008, the applicant responded to a request for information from the staff (RAI-TR142-SPCV-02 through RAI-TR142-SPCV-04). The applicant provided the results of an analysis demonstrating that most of the inlet area would have to be blocked before the PCS function is degraded. In addition, the applicant stated the following:

> There is a possibility that these [inlet] screens could become clogged with airborne debris. For this reason, there is access to the louvers and screens by an enclosed walkway between the wall containing the louvers and the shield building wall. Regular inspections of the louvers will be made and the screens will be kept free of debris. The frequency of inspections is expected to be once per month, and may change depending on the degree of blockage observed.

The staff finds that this commitment and the surveillance requirement (SR) provide assurance that significant air blockages will not exist before high winds or ice storms. Because of the very large degree of blockage that would be required to challenge containment cooling, the elevation of the openings, and the design of the airflow path, the staff considers the frequency of such an event to be negligible.

In DCD Revision 17, insights related to the protected location of DAS manual actuation cables and the assurance of adequate containment cooling air flow have been added to Table 19.59-18. The staff considers the portions of RAI-SRP19.0-SPLA-01 related to high winds to be resolved.

19.1.5.5 External Flooding Evaluation

The applicant assessed various scenarios with the potential to raise water levels and concluded that, even in combination, external flooding of safety-related SSCs (and certain risk-significant investment protection equipment) is prevented. For that reason, the effect of external flooding on risk-significant SSCs was not evaluated. Although storms, dam failure, and other external phenomena can cause flooding, the analysis assumed that the generic site was not subject to flooding by flash floods or failure of an upstream dam. COL applicants must show that this is also true for their proposed site. Otherwise, COL applicants must evaluate the other external events that can raise water levels at the site.

The risk evaluation of external floods is based on the following features from the design basis of the plant documented in Section 2 of the AP1000 DCD:

- The AP1000 is protected against floods up to the 30.5 m (100-ft) level, which corresponds to the plant ground level. From this point, the ground is graded so that water naturally flows away from the plant structures.

- The plant is designed such that the 30.5 m (100-ft) level is slightly above grade and the level of anticipated external flooding. Below grade is protected against flooding by a water barrier consisting of waterstops and a waterproofing system. Seismic Category I SSCs below grade are designed to withstand hydrostatic pressures.

- The seismic Category I SSCs below grade (below ground level) are protected against flooding by a water barrier consisting of waterstops and a waterproofing system.

In RAI-SRP19.0-SPLA-01, the staff also requested that these insights related to external flooding be added to the DCD (RAI-SRP19.0-SPLA-01 also discussed high winds cooling, addressed in Section 19.1.5.4).

In a letter dated July 22, 2008, the applicant proposed a revision to DCD Table 19.59-18 identifying these features and requirements among the PRA assumptions and insights. This is

consistent with NUREG-0800 and is, therefore, acceptable to the staff. In DCD Revision 17, insights related to external flooding have been added to Table 19.59-18. The staff considers the portions of RAI-SRP19.0-SPLA-01 related to external flooding to be resolved.

The risk evaluation of external floods is highly specific to the proposed plant site. A detailed risk analysis of external floods requires information not only on potential sources of water but also on local configurations such as dikes, surface grading, locations of structures, and location of equipment within the structures. However, bounding assumptions about candidate sites are used to show that external floods can be screened from detailed risk evaluation. Assuming that the ground is graded away from the structures and there is no site susceptibility to dam failure or flash flooding, the remaining source of external flooding is storm surges capable of reaching the plant ground level. Hurricanes can produce the highest storm surges. The applicant performed a screening risk evaluation using as a reference the site most susceptible to external floods from hurricane surge water among potential candidate sites for an AP1000 plant. This site is located at an elevation of 13.7 m (45 ft) above sea level and in an area where the highest storm surges due to hurricanes have occurred. All other proposed sites are located at higher elevations above sea level. Therefore, it would require a 13.7 m (45-ft) hurricane storm surge to reach the plant ground level. Any surge that stops below ground level at the plant has no impact on the plant due to flooding. Based on the Saffir-Simpson hurricane scale (Table 19.58-2 of the AP1000 DCD), only Category 5 hurricanes have the ability to generate storm surges in excess of 5.49 m (18 ft). Historically, the highest observed storm surges occurred during hurricane Katrina in 2005 and hurricane Camille in 1969. The maximum high water mark observation occurred during hurricane Katrina with a surge of 8.47 m (27.8 ft) above normal tide levels at Pass Christian, on the immediate Gulf Coast just east of St. Louis Bay.

Based on the historical information, documented in Section 19.58.2.2 of the DCD, the applicant stated that a hurricane storm surge in excess of 8.53 m (28 ft) can be classified as a rare event and a hurricane storm surge in excess of 13.7 m (45 ft) as an extremely rare event and can be assigned a frequency of 1×10^{-7} per year or less. In addition, a risk assessment that was performed as a sensitivity study, which assumed loss of the switchyard and all nonsafety-related SSCs, indicated that the CDF associated with external floods is insignificant when areas containing safety-related equipment are protected. Therefore, by recognizing the fact that the AP1000 design provides features (e.g., barriers) that provide protection against the propagation of flooding to areas where safety-related equipment is located, external floods that do not closely approach ground level at the plant are screened from detailed risk evaluation in accordance with the criteria discussed in Section 19.1.5.

On the basis of the discussion in DCD Section 19.58 and the large margin between the greatest storm surge observed and the water level required to affect plant safety, the staff finds that the frequency of external flooding caused by storm surge is negligibly small. The screening criteria are met and no further analysis is required. The staff expects a COL applicant to verify the applicability of the screening criteria to the proposed site.

COL applicants must confirm that all possible mechanisms of external flooding at the proposed site have been assessed, including credible combinations of those mechanisms. Otherwise, the COL applicant must screen out these external events by demonstrating that they occur with negligible frequency. COL Information Item 19.59.10-2 requires the COL applicant to re-evaluate the qualitative screening of external events.

Because this approach is consistent with RG 1.200, the staff considers this an acceptable way to address the risk of external flooding.

19.1.5.6 Transportation and Nearby Facilities Accident Evaluation

Section 19.58.2.3 of the AP1000 DCD discusses the risk from external events related to transportation accidents near the nuclear plant and to accidents at nearby industrial and military facilities. DCD Section 19.58 discusses the following types of accidents: (1) aviation, (2) marine, (3) pipeline, and (4) railroad and truck. The staff finds that these accidents form an acceptable set of transportation and nearby facility accidents based on the modified IPEEE guidelines discussed in Section 19.1.5, above. Each COL applicant should verify that these analyses bound all such hazards relevant to the proposed site.

19.1.5.6.1 Aviation Accidents

The risk evaluation for aviation accidents considers two cases: (1) small aircraft impact; and (2) commercial aircraft impact. The applicant screened small aircraft impact accidents from detailed risk evaluation by performing a bounding risk evaluation based on a limiting frequency of 1.2×10^{-6} impact events per year.

For small aircraft impact accidents, the applicant performed a bounding risk evaluation that assumed a limiting initiating event frequency of 1.2×10^{-6} per year, together with an LOSP and loss of nonsafety systems. The likelihood that the impact of a small aircraft would challenge the safety systems (all located within the nuclear island) is considered negligible. Assuming that a small aircraft impact could cause an LOSP and loss of nonsafety systems, the applicant showed that the associated CDF is less than 1×10^{-8} per year.

The applicant screened commercial-size aircraft impact accidents from detailed risk evaluation by assuming a limiting frequency of 1×10^{-7} impact events per year. Each COL applicant will demonstrate the assumed limiting event frequency for the selected site of 1.2×10^{-6} per year for small aircraft and 1×10^{-7} per year for commercial-size aircraft.

19.1.5.6.2 Marine Accidents

Sites close to large waterways with ship and/or barge traffic need to evaluate the risk associated with marine accidents. Marine accidents pose a hazard to a nuclear power plant due to: (1) release of hazardous material towards the plant; and (2) explosion with resulting damage to the plant.

The applicant evaluated the risk associated with the release of hazardous material towards the plant, following a marine accident, using a bounding analysis that assumed a limiting initiating event frequency of 1×10^{-6} per year, and showed that the associated CDF is less than 1×10^{-8} per year. Therefore, based on the screening criteria discussed in Section 19.1.5, the release of hazardous material in a marine accident requires no detailed risk evaluation. The applicant modeled the risk impact of a toxic release by assuming a reactor trip and guaranteed failure of all operator actions credited in the PRA (the toxic release is not expected to lead to any direct failure of safety equipment). This is a conservative analysis because the AP1000 has an additional level of defense against toxic airborne material. Specifically, with warning that a release has occurred, the operators can actuate passive control room habitability. This system isolates the control room from normal heating, ventilation, and air conditioning (HVAC) and actuates a separate system supplied from compressed air containers. The compressed air slightly pressurizes the control room above atmospheric pressure, preventing the entrance of toxic material for at least 72 hours. (This is adequate time for operators to deal with the event.)

The staff review finds that the bounding evaluation, documented in Section 19.58.2.3.2 of the AP1000 DCD, demonstrates that the risk associated with the release of hazardous materials in a marine accident is insignificant, assuming a limiting initiating event frequency of 1×10^{-6} per year. The COL applicant for the selected site will demonstrate the assumed frequency of 1×10^{-6} per year for release of hazardous materials that could pose a hazard to the plant by a marine accident.

The applicant screened out qualitatively the risk associated with an explosion following a marine accident with resulting damage to the plant from a detailed analysis based on the acceptance criteria of event frequency less than 1×10^{-7} per year and CDF less than 1×10^{-8} per year for the following reasons:

- Loss of service water events resulting from a marine explosion is not a nuclear safety concern for AP1000 since the design does not include a service water intake structure.

- RG 1.91 provides the acceptance criterion of an overpressure event in excess of 6.9 kilopascal (kPa) (1 pound per square inch (psi)) at a frequency of less than 1×10^{-6} per year.

- Margin above the RG 1.91 acceptance criterion has been demonstrated. A study for the Waterford site, NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States: Other External Events," Supplement 2, indicated that the AP1000 safety-related buildings can withstand overpressures above the RG 1.91 acceptance criterion of 6.9 kPa (1 psi).

The staff review finds that the risk associated with an explosion following a marine accident is insignificant when the RG 1.91 acceptance criterion of an overpressure event in excess of 1 psi at a frequency of less than 1×10^{-6} per year is met. The COL applicant will demonstrate that the RG 1.91 criterion of an overpressure event in excess of 1 psi at a frequency of less than 1×10^{-6} per year is met for the selected site.

19.1.5.6.3 Pipeline Accidents

Sites close to pipelines need to evaluate the risk associated with pipeline accidents. Pipeline accidents pose a hazard to a nuclear power plant because of the potential for: (1) a release of hazardous material towards the plant; and (2) an explosion with resulting damage to the plant. The applicant evaluated the risk associated with the release of hazardous material towards the plant following a pipeline accident using a bounding analysis that assumed a limiting initiating event frequency of 1×10^{-6} per year. This analysis showed that the associated CDF is less than 1×10^{-8} per year. Therefore, based on the screening criteria discussed in Section 19.1.5, the release of hazardous material in a pipeline accident requires no detailed risk evaluation. The applicant modeled the risk impact of a toxic release by assuming a reactor trip and guaranteed failure of all operator actions credited in the PRA (the toxic release is not expected to lead to any direct failure of safety equipment). This is a conservative analysis because the AP1000 has an additional level of defense against toxic airborne material. Specifically, with a warning that a release has occurred, the operators can actuate passive control room habitability. This system isolates the control room from normal HVAC and actuates a separate system supplied from compressed air containers. The compressed air slightly pressurizes the control room above atmospheric pressure, preventing the entrance of toxic material for 72 hours. (This is adequate time for operators to deal with the event.)

The risk associated with an explosion following a pipeline accident with resulting damage to the plant was screened out qualitatively from a detailed analysis, based on the acceptance criteria of event frequency less than 1×10^{-7} per year. Section 19.58.2.3.3 of the AP1000 DCD documents an approach (pipeline accident model) that qualitatively illustrates potential scenarios from gas pipeline accidents. This approach briefly discusses the following considerations for evaluating the frequency of pipeline accidents: (1) gas pipe rupture frequency estimation; (2) gas cloud formation probability estimation; (3) gas cloud transportation and nondispersion probability estimation; and (4) onsite gas cloud ignition probability estimation.

The staff review finds that the risk associated with pipeline accidents is insignificant, assuming that the COL applicant will demonstrate that the frequency criterion of 1×10^{-7} per year is met for the proposed site for pipeline accidents that could pose a hazard to the plant. The frequency of pipeline accidents will be evaluated by the COL applicant using the approach discussed in Section 19.58.2.3.3 of the DCD or another approach acceptable to the staff.

19.1.5.6.4 Railroad and Truck Accidents

Railroad and truck accidents could pose a hazard to an AP1000 plant, and COL applicants need to evaluate the risk associated with such accidents. As for marine and pipeline accidents, railroad and truck accidents could pose a hazard to a nuclear power plant because of the potential for: (1) a release of hazardous material towards the plant; and (2) an explosion with resulting damage to the plant. However, railroad and truck accidents are expected to be less likely to occur (e.g., because of the improved security barriers established at U.S. nuclear power plants) and cause less plant damage than aviation or marine accidents if they should happen. For these reasons, the risk impact from railroad and truck accidents is insignificant if the initiating event frequency criterion of 1×10^{-7} per year is met.

The staff review finds that the risk associated with railroad and truck accidents is insignificant, assuming that the COL applicant can demonstrate that the frequency criterion of 1×10^{-7} per year is met for the proposed site for railroad and truck accidents that could pose a hazard to the plant.

19.1.5.7 External Fires

External fires are those that occur outside the controlled site boundary. Potential effects on the plant could be LOSP, forced isolation of the plant ventilation, and control room evacuation. External fires are not expected to spread on site because of site clearing during the construction phase and control of combustibles during construction and operation. In RAI-TR101SPLA-03, the staff requested that the applicant consider external fires more explicitly.

In a letter dated February 8, 2008, the applicant agreed to address that based on site-specific information, the COL applicant should reevaluate the qualitative screening of external fires. Accordingly, based on the criteria discussed in Section 19.1.5, above, which were used to screen out external hazards in the PRA, a risk evaluation should be performed if the COL applicant cannot demonstrate that the frequency of external fires that could pose a hazard to the plant is less than 1×10^{-7} per year. If the COL applicant identifies any site-specific susceptibilities, the site-specific PRA performed by licensees to address COL Information Item 19.59.10-2 should include external fires.

This is consistent with RG 1.200 and, therefore, acceptable to the staff. The staff considers RAI-TR101SPLA-03 resolved.

19.1.5.8 Conclusions

Information documented in Section 19.58 of the AP1000 DCD addresses the second part of COL Information Item 19.59.10-2 which reads as follows:

> Based on site-specific information, the COL should also re-evaluate the qualitative screening of external events.... If any site-specific susceptibilities are found, the PRA should be updated to include the applicable external event.

The information provided in Section 19.58 of the AP1000 DCD includes the following objectives:

- to show that screening criteria are met and to identify external events that may impact the AP1000 risk on a site-specific basis
- to provide generic risk analyses, based on bounding assumptions regarding site-specific parameters (e.g., frequency of each category of hurricanes) for some external events
- to provide guidance to COL applicants regarding the verification of the applicability of these "generic" analyses to a specific site

Based on modified IPEEE guidelines, DCD Section 19.58 discusses the following external events or external hazards:

- high winds (including tornadoes)
- external floods
- transportation and nearby facility accidents
- external fires

The staff review finds that these external hazards are most likely a complete list of events associated with candidate sites for an AP1000 plant. However, as stated in DCD Section 2.2.1, "Combined License Information for Identification of Site-specific Potential Hazards," the COL applicant must verify that this list adequately addresses external hazards at the proposed site. The COL applicant should use site-specific information to verify that the assumptions made in the analyses performed during the DC stage are applicable. For example, screening on the basis of event frequency of external flooding due to tsunamis or upstream dam failures may not be possible at all sites.

The staff finds that the generic risk analyses and other information provided in Section 19.58 of the AP1000 DCD are acceptable, including the screening of events from inclusion in the PRA, given the documented assumptions.

However, these analyses are based on assumptions that are expected to envelop site-specific information at sites selected to build a nuclear plant referencing the AP1000 design. The COL applicant referencing the information provided in DCD Section 19.58 must: (1) confirm in the COL application that the information provided in Section 19.58 of the DCD is applicable to the selected site; and (2) ensure that the assumptions made in the generic risk evaluations documented in Section 19.58 of the DCD bound the site-specific conditions for the applicant's

site. This is in agreement with the stipulation made in Section 19.58.3 of the AP1000 DCD, which states that the COL applicant should conduct a site-specific review of the generic PRA to verify that the assumptions in the PRA bound the site-specific conditions for the applicant's site (COL Information Item 19.59.10-2).

19.1.8.24 Reactor Pressure Vessel Thermal Insulation System

The AP1000 design includes a reflective RVIS that provides an engineered flow path to allow water ingress and venting of steam for external reactor vessel cooling (ERVC) in the event of a severe accident involving core relocation to the lower plenum. COL Action Item 19.2.3.3.1.3.2-1 calls for the COL applicants to complete the design for the RPV thermal insulation system. Section 39.10.2 of the AP1000 PRA specifies its functional requirements. In addition to RCS depressurization and reactor cavity flooding, several other conditions are necessary: (1) the reactor vessel thermal insulation system design must be consistent with ULPU Configuration V testing with prototypical insulation (ULPU-2000 is a boiling heat transfer test facility at the University of California at Santa Barbara used to investigate in-vessel retention of a damaged core); (2) the RVIS must maintain its integrity under the hydrodynamic loads associated with ERVC and not be subject to clogging of the coolant flow path by debris; and (3) RPV exterior coatings do not preclude the wetting phenomena identified as the cooling mechanism in the ULPU testing.

The applicant has completed the design of the RVIS. In TR-24, the applicant provided information to demonstrate that the RVIS is designed to provide adequate cooling to ensure in-vessel retention of a damaged and relocated core. On this basis, the applicant proposed to close COL Information Item 5.3-5. The staff's evaluation is in Section 19.2.3.3.1.3.2 of this report.

19.1.9 Conclusions and Findings

The staff has evaluated the AP1000 design PRA quality and its use in the design and certification processes. The NRC concludes that the quality and completeness of the AP1000 PRA are adequate for its intended purposes, which are to support the design and certification processes and satisfy the requirements of 10 CFR 52.47. The approaches used by the applicant for both the core damage and containment analyses are logical and sufficient to achieve the desired goals of describing and quantifying potential core damage scenarios and containment performance during severe accidents.

The use of PRA in the AP1000 design process improved the unique passive features of the design by providing a better understanding of plant response, including potential system interactions, during postulated accidents beyond the design basis. Such features contributed to the reduced CDF and conditional containment failure probability estimates of the AP1000 design when compared to those of operating PWRs. The applicant used the PRA results and insights to identify areas in which it is particularly important to implement the certification and operational requirements assumed during the design and certification processes (e.g., ITAAC, RTNSS requirements, D-RAP, COL information items, and technical specifications).

The staff reviewed the description and results of the PRA provided in the amendment. The staff finds that they are consistent with the requirements of 10 CFR 52.47 and the guidance provided in RG 1.200, NUREG-0800 Sections 19.0 and 19.1, as well as DC/COL-ISG-1, DC/COL-ISG-3 and DC/COL-ISG-20. Therefore, the staff concludes that the amended AP1000 design meets

the NRC's safety goals and represents an improvement in safety over current operating PWRs in the United States.

19.1.10 Resolution of Safety Evaluation Report Open Items

Open Item OI-SRP19.0-SPLA-07: The applicant must implement corrective actions after the audit, re-solving and requantifying the corrected model as well as revising TR-102 and making associated changes to the DCD consistent with DC/COL-ISG-3.

As discussed in Section 19.1.2.1.10, "Improved Control Room Design and Digital Instrumentation and Control Systems," this open item has been resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Open Item OI-SRP19.0-SPLA-12: The applicant must confirm that an acceptable seismic margin is maintained for HRHF sites.

As discussed in Section 19.1.5, "Seismic Margin Analysis," this open item has been resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Open Item OI-SRP19.0-SPLA-13: The applicant must provide an updated DCD description of events (HAs, common cause, and basic) and sequences contributing most to risk, both at power and while shutdown.

As discussed in Section 19.1.4, "Safety Insights from the Internal Events Risk Analysis for Shutdown Operation," this open item has been resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Open Item OI-SRP19.0-SPLA-14: The applicant must resolve the discrepancy in the containment inventory of radionuclides used for the survivability evaluation, determining whether the environmental assessment should include mechanical penetrations and hatches (e.g., gasket materials) and providing a licensee COL information item to finalize the list of equipment that must survive.

As discussed in Section 19.2.3.3.7, "Equipment Survivability," this open item has been resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Open Item OI-SRP19F-SPLA-01: In a letter dated March 19, 2010, the applicant submitted the results of an analysis consistent with guidance from the Nuclear Energy Institute. It was prepared using NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," which has been endorsed in draft guide (DG)-1176, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts." The staff's evaluation of this aircraft impact assessment (AIA) is in Section 19F of this report.

19.1.11 Combined License Information Items

19.1.11.1 As-Built Seismic Margin Assessment

COL Information Item 19.59.10-1 (COL Action Items 19A.2-1 and 19A.2-2) is associated with an as-built SSC HCLPF comparison to seismic margin evaluation. In TR-6, the applicant noted that a COL applicant cannot complete the review and proposes instead to complete these actions before fuel loading.

However, at the time of COL application, site-specific features with the potential to affect the SMA should be described and assessed to confirm that the SMA documented in the DCD is applicable to a plant located on the proposed site. Alternatively, additional analysis to demonstrate adequate seismic margin is required. COL Information Item 19.5.10-6 is a COL applicant item.

Because the SMA requires a confirmatory walkdown, which cannot be performed until construction of seismic SSCs has been completed, the staff agrees that evaluation of as-built conditions cannot be provided with the COL application. The staff concludes that performance of the as-built SMA before fuel loading is timely and, therefore, acceptable. COL Information Item 19.59.10-1 should be completed by the licensee.

19.1.11.2 Site-Specific, Plant-Specific Probabilistic Risk Assessment

COL Information Item 19.59.10-2 is associated with evaluating an as-built plant versus design in AP1000 PRA and site-specific PRA external events. In TR-6, the applicant noted that a COL applicant cannot complete an as-built review and proposes that the COL applicant referencing the AP1000 certified design will, instead, review differences between the as-built plant and the design used as the basis for the AP1000 PRA and Table 19.59-18. The applicant proposed that, if a screening analysis shows that the effect of the differences could result in a significant increase in CDF or LRF, the PRA would be updated to reflect these differences. In addition, the COL applicant should reevaluate the qualitative screening of external events. If any site-specific susceptibilities are found, the PRA should be updated to include the applicable external event.

The staff agrees that the design-specific PRA is a sufficient and acceptable basis for drawing safety conclusions for a license and should be described in the COL FSAR (with appropriate discussion of plant-specific features and departures from the certified design). The FSAR should also identify key assumptions and insights from this PRA. The staff finds that documentation and qualitative screening of external events, specific to the proposed plant site, is an acceptable way to confirm that site-specific vulnerabilities do not require further risk assessment and need not be included in PRA results and insights reported in the FSAR. This part of COL Information Item 19.5.10-2 is a COL applicant item.

However, the staff finds that each licensee's PRA should model significant plant-specific and site-specific differences from the design PRA, whether positive or negative, to be consistent with DC/COL-ISG-3. This is necessary to support operational-phase reliability assurance activities, when more realistic assessment of risk is needed to avoid masking activities of risk significance. This part of COL Information Item 19.5.10-2 is to be completed by the COL licensee.

19.2 Severe Accident Performance

19.2.2 Deterministic Assessment of Severe Accident Prevention

19.2.2.1.2 Mid-Loop Operation

During refueling or maintenance activities, the RCS is sometimes reduced to a "mid-loop" level. The applicant summarized the specific AP1000 design features that address mid-loop operations in DCD Tier 2, Section 5.4.7.2.1, "Design Features Addressing Shutdown and Mid-Loop Operations." In addition, DCD Tier 2, Table 16.3-2, "Investment Protection Short-Term Availability Controls," ensures that the RNS is available during mid-loop operation.

Section 19.3, "Shutdown Evaluation," of this report documents the staff's evaluation of shutdown risk. Because RTNSS has not been amended, it is not discussed in this supplement. DCD Tier 2, Table 16.3-2, documents the availability controls provided for the RNS during normal and reduced inventory. The staff concludes that the AP1000 design conforms to the mid-loop operation guidance specified in SECY-93-087 and is, therefore, acceptable.

19.2.3.3.1.3.2 Reactor Pressure Vessel Thermal Insulation System

Section 5.3.5 describes the design of the RPV thermal insulation system. Section 19.1.2.2.4 discusses considerations related to risk assessment. This section addresses the staff's evaluation of conformance of the final design of the RVIS to the ULPU Configuration V testing, its integrity under the hydrodynamic loads associated with ERVC, absence of susceptibility to clogging, and absence of coatings that could interfere with wetting phenomena that contribute to effective heat removal.

In RAI-TR24-SPLA-01, the staff noted an apparent stepwise change in the cross-section of the annulus formed between the RVIS and the reactor vessel, formed by the neutron shield. This RAI also requested other information with respect to dimensions that could be important to the adequacy of the flow path. In a letter dated August 21, 2007, the applicant provided additional details about the modeling of the flow path and flow areas. This allowed the staff to confirm that the design was consistent with the ULPU Configuration V testing. The staff considers RAI-TR24-SPLA-01 resolved.

The applicant changed the design of the RVIS inlet closure devices from floating balls to hinged, buoyant doors. In RAI-TR24-SPLA-02, Part 1, the staff requested additional information to confirm that this was consistent with the ULPU Configuration V testing. In RAI-TR24-SPLA-07, the staff requested additional details on the configuration of the doors and the forces acting on them. In the August 21, 2007 letter, the applicant provided additional details about these active components of the system and the effect on flow areas and flow resistance. Because the new design retains the characteristic of actuation by buoyant forces, the cross-sectional area is maintained, and flow resistance is reduced, the staff considers this change to be an acceptable way to conform to the ULPU Configuration V testing.

Similarly, the applicant changed the design of steam vent ducts that provide a flow path for the steam/water within the reactor vessel insulation annular space to flow back to the containment flood-up region. In RAI-TR24-SPLA-02, Part 2, the staff requested that the applicant assess the impact of these design modifications. In the August 21, 2007 letter, the applicant explained that the previous design had multiple miter bends instead of a sudden contraction in the area of the

flow path. The modification reduces flow resistance and allows higher mass flow. Because the results of the testing conservatively bound the expected performance of the RVIS, the staff considers this change to be an acceptable way to conform to the ULPU Configuration V testing. The staff considers RAI-TR24-SPLA-02 and RAI-TR24-SPLA-07 resolved.

The staff noted that the RVIS doors and the reactor coolant drain tank room ventilation damper, though described by the applicant as "passive," are considered by the staff to be active components. In RAI-TR24-SPLA-03, the staff requested information on periodic verification of the performance of moving parts, and in RAI-TR24-SPLA-08, a discussion of as low as is reasonably achievable (ALARA) considerations for testing and maintenance was requested. In the August 21, 2007 letter, the applicant confirmed that RVIS is in the D-RAP. Proper fit and freedom of motion of the doors is confirmed during hot functional testing. Visual inspection and testing for freedom of rotation is to be performed during refueling outages at ten-year intervals, coordinated with other inspections in the same area. Individual doors and frames are designed for removal as a unit, so replacement, if required, would take little time. Because the applicant has applied a level of control and testing consistent with Commission policy on RTNSS and ALARA, the staff considers this to be acceptable and both RAI-TR24-SPLA-03 and RAI-TR24-SPLA-08 are resolved.

Because the design of the RVIS is consistent with the ULPU Configuration V testing, the staff's evaluation of hydrodynamic loads and previous findings with respect to the potential for clogging of the flow path are unchanged.

In RAI-TR24-SPLA-06, the staff requested resolution of earlier test results (ULPU Configuration III and BETA tests) dealing with the wetability of the reactor pressure vessel surface if it were coated. The applicant revised the DCD to reflect a commitment to ensure that the reactor vessel exterior is bare metal.

On the basis of the additional description, the staff was able to confirm that the new design is consistent with the ULPU Configuration V ntesting and is, therefore, acceptable. The staff concludes that COL Action Item 19.2.3.3.1.3.2-1 is closed for COL applicants referencing the AP1000 DCD. The staff considers RAI-TR24-SPLA-06 resolved.

19.2.3.3.7 Equipment Survivability

Electrical and mechanical equipment must survive to prevent and mitigate the consequences of severe accidents. The applicant addressed equipment survivability in Appendix 19D, "Equipment Survivability Assessment," to DCD Tier 2, which includes general requirements and equipment classification. Appendix D to the AP1000 PRA supporting document presents the analysis performed to determine the severe accident environmental conditions.

In TR-68, APP-GW-GLR-069, "Equipment Survivability Assessment," May 2007, the applicant submitted revised analysis in support of the DC amendment. In reviewing TR-68, the staff noted that the severe accident environmental conditions were revised.

In APP-GW-VP-025, "AP1000 Equipment Survivability Assessment," an attachment to TR-68, the applicant stated that it had revised the fraction of the core inventory released to the containment atmosphere from the original PRA Appendix D and that the values are consistent with the accident source term information presented in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants—Final Report."

However, the staff identified certain analytical assumptions documented in APP-GW-VP-025 that did not appear to be consistent with NUREG-1465 and asked the applicant to clarify and confirm the basis for the results documented in TR-68. The NRC staff identified this as the first part of Open Item OI-SRP19.0-SPLA-14.

In a letter dated May 12, 2009, the applicant submitted additional information to resolve the perceived lack of consistency, and supplemented this with a letter dated September 8, 2009. The staff reviewed supporting calculations during an audit at the applicant's offices on October 13-15, 2009, documented in an audit report dated November 22, 2009. The staff is satisfied that the post-accident conditions were determined in a manner consistent with the applicable regulatory guidance, and the first of three parts of Open Item OI-SRP19.0-SPLA-14 is considered to be closed.

During the development of the severe accident management guidance (SAMG) for AP1000, additional requirements were defined for accident management in Time Frame 2 (in-vessel severe accident phase) and Time Frame 3 (ex-vessel severe accident phase). For example, previously unidentified methods of injecting water into the containment were added (e.g., providing makeup to overflow the IRWST by the RNS system). The use of containment spray was identified as a severe accident strategy (e.g., injecting water into containment and containment heat removal). Another method of depressurizing the RCS in Time Frame 2 was identified (i.e., reactor vessel head venting).

Finalizing certain system designs resulted in the need to update lists of associated equipment and instrumentation. For example, the applicant eliminated low-pressure steam generator feed systems (i.e., service water and condensate water) from consideration.

Lastly, the applicant changed the equipment and instrumentation identification to conform to updated naming conventions for AP1000. The new list of equipment and instrumentation reflects the amended AP1000 design.

The applicant has not completed the identification of equipment and instrumentation for prevention of core damage (e.g., Time Frame 0 and Time Frame 1) because the EOPs are still in development. Upon finalization of the EOPs, the applicant can identify and assess the survivability of the equipment and instrumentation used in those procedures. This should be identified as a licensee COL information item. The NRC staff identified this as the second part of Open Item OI-SRP19.0-SPLA-14.

In a letter dated May 12, 2009, the applicant submitted additional information that clarified the method by which this licensee COL information item is addressed. The staff finds that no additional holder item is required and, therefore, the second of three parts of Open Item OI-SRP19.0-SPLA-14 is considered to be closed. In general, the applicant claims that the AP1000 provides reasonable assurance that equipment, both electrical and mechanical, designed for mitigating the consequences of severe accidents will perform its functions as intended.

The staff questioned the completeness of the list of SSCs required for containment isolation. First, if a hydrogen monitor outside containment were required, additional penetrations might need to be included. In addition, it was not clear that the equipment survivability assessment needs to include mechanical penetrations and hatches (e.g., gasket materials) to ensure containment integrity. The NRC staff identified this as the third part of Open Item OI-SRP19.0-SPLA-14.

In a letter dated May 12, 2009, the applicant submitted additional information, and supplemented this with a letter dated September 8, 2009. The staff is satisfied that a hydrogen monitor outside containment was not included in the certified design and finds that there is no regulatory basis for requiring it. The applicant provided appropriate controls to ensure that gasket materials of mechanical penetrations and hatches will be capable of surviving post-accident conditions. In addition, the applicant has proposed an acceptable clarification of the DCD. The third of three parts of Open Item OI-SRP19.0-SPLA-14 is considered resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19.2.5 Accident Management

Accident management encompasses those actions taken during the course of an accident by the plant operating and technical staff to: (1) prevent core damage; (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel; (3) maintain containment integrity as long as possible; and (4) minimize offsite releases. Severe accident management is the process of extending the EOPs well beyond the plant design basis into severe fuel damage regimes and making full use of existing plant equipment, operator skills, and creativity to terminate severe accidents and limit offsite releases.

The NRC has taken an active role in ensuring that utilities adopt acceptable accident management practices. In January 1989, the staff issued SECY-89-012, "Staff Plans for Accident Management Regulatory and Research Programs," which discusses essential elements of a utility accident management plan and offers an approach for accident management implementation. Subsequently, the NRC worked with the industry to define the scope and attributes of a utility accident management plan and to develop guidelines for plant-specific implementation. Section 5 of NEI 91-04, Revision 1, "Severe Accident Closure Guidelines," which lays out the elements of the industry's severe accident management closure actions that have been accepted by the staff, resulted from these efforts. This program involves the development of: (1) a structured method by which utilities may systematically evaluate and enhance their abilities to deal with potential severe accidents; (2) vendor-specific accident management guidelines for use by individual utilities in establishing plant-specific accident management procedures and guidance; and (3) guidance and material to support utility activities related to training in severe accidents. Using the guidance developed through this program, each operating plant has implemented a plant-specific accident management plan as part of an industry initiative.

Based on its reviews of these efforts, severe accident evaluations in individual plant examinations, and industry PRAs, the staff concluded that improvements to utility accident management capabilities could further reduce the risk associated with severe accidents. Although new reactor designs are to have enhanced capabilities for the prevention and mitigation of severe accidents, accident management remains an important element of defense in depth for these designs. However, the staff expects the increased attention on accident prevention and mitigation in these designs to alter the scope and focus of accident management relative to that for operating reactors. For example, the staff expects increased attention on accident prevention and the development of error-tolerant designs to decrease the need for operator intervention, while increasing the time available for such action if necessary. This permits a greater reliance on support from outside sources. For longer times after an accident (several hours to several days), the need for human intervention and accident management will continue.

For both operating and advanced reactors, the overall responsibility for accident management, including development, implementation, and maintenance of the accident management plan, lies with the nuclear utility, because the utility bears ultimate responsibility for the safety of the plant and for establishing and maintaining an emergency response organization capable of effectively responding to potential accident situations. However, the vendors have played key roles in providing essential SAMG and strategies for implementation. This guidance has served as the basis for severe accident management procedures and for training utility personnel in carrying out the procedures. Computational aids for technical support have been developed, information needed to respond to a spectrum of severe accidents has been provided, decision making responsibilities have been delineated, and utility self-evaluation methodologies have been developed.

A COL applicant referencing the AP1000 design must develop and implement SAMG using the suggested framework provided in WCAP-13914, "Framework for AP600 Severe Accident Management Guidance," Revision 3. This WCAP outlines a plan based on the Westinghouse Owners Group (WOG) SAMG for currently operating plants. Its scope is to address significant core damage accidents that might be possible in the AP600 and to provide the framework for developing guidance on how to cope with these accidents after the emergency response guidelines are no longer applicable. TR-66 extends this framework to the AP1000 design.

In NUREG-1793, the staff states that it expects the COL applicant to follow the recommendations provided in WCAP-13914 in developing its plant-specific accident management guidance (COL Action Item 19.2.5-1). The applicant has taken steps to facilitate this process by producing TR-66, the AP1000 framework document, and the AP1000 SAMG.

The applicant prepared and submitted TR-66 to close COL Information Item 19.59.10-4 with respect to development of the SAMG. The information item states the following:

> The combined license applicant referencing the AP1000 certified design will develop and implement severe accident management guidance using the suggested framework provided in WCAP-13914, "Framework for AP600 Severe Accident Management Guidance."

APP-GW-GL-027, "Framework for AP1000 Severe Accident Management Guidance Development," documents the framework for the AP1000. Based on this framework, the applicant has also developed SAMG for the AP1000 (APP-GW-GJR-400, "AP1000 Severe Accident Management Guidelines," Revision A of January 2007), which COL licensees will implement at each site using the AP1000 design. TR-66 is a road map for COL licensees using these guidelines.

In RAI-SRP19.0-SPLA-15, the staff requested clarification of the schedule for development and implementation of SAMG. In a letter dated August 21, 2008, the applicant provided the requested information.

The starting point for the technical basis for the AP1000 is Electric Power Research Institute (EPRI) TR-101869, "Severe Accident Management Technical Basis Document" (Volumes 1 and 2). This document details the severe accident phenomenological understanding as it was in the early 1990s, when it formed the basis for the WOG guidelines. For the most part, this understanding is still current, although a number of important technical issues have been resolved and a few new ones identified since then. For example, direct containment heating in

large-volume PWRs is no longer considered to be a major threat, but induced SGTRs in high-pressure scenarios have become a major concern. The AP1000 SAMG consists of three volumes:

1. An executive volume describes the methodology and criteria for the development of the AP1000 SAMG. It includes all of the material in the framework document, an overview of the AP1000 SAMG, a writer's guide for writing the SAMG and background documents, and a number of other important items related to the decision making process, interfaces between the AP1000 EOPs and the SAMG, and interfaces between the SAMG and the site emergency plan.

2. A guideline volume includes the SAMG guidelines to be used by the control room staff and the engineering support staff in the technical support center (TSC) in responding to a severe accident.

3. A background volume details the technical basis for the guidance found in the guidelines volume.

Section 5.1 of NEI 91-04 states that accident management consists of those actions taken during the course of an accident by the plant's emergency response organization (ERO), particularly the plant operations, technical support, and plant management staff, to achieve the following:

- prevent the accident from progressing to core damage
- terminate core damage progression once it begins
- maintain the capability of the containment as long as possible
- minimize onsite and offsite releases and their effects

The latter three actions constitute a subset of accident management referred to as severe accident management or, more specifically, severe accident mitigation.

NEI 91-04 also states that the goal of severe accident management is to enhance the capabilities of the ERO to mitigate severe accidents and prevent or minimize any offsite releases. The objective is to establish core cooling and to manage any current or immediate threats to the fission product barriers. Accomplishing this ERO should make full use of existing plant capabilities, including standard and nonstandard uses of plant systems and equipment.

The NRC staff agrees that NEI 91-04 properly defines the scope of severe accident management and believes that the framework for SAMG should be consistent with this scope.

The AP1000 SAMG consists of three parts: control room SAMG, TSC SAMG, and TSC challenge response guidance. The control room SAMG consists of two separate guidelines. The control room staff uses the first of these guidelines until the TSC is functional and its staff is ready to use the TSC SAMG. The staff uses the second guideline after the TSC is functional; this guideline provides the staff with a structured set of activities when the TSC is evaluating the plant conditions and potential responses.

The TSC staff will execute the TSC SAMG, using the diagnostic flow chart (DFC) and the severe challenge status tree to select the appropriate strategies to respond to variations in the key parameters. These strategies are described by the seven severe accident guidelines:

1. inject into containment
2. depressurize the RCS
3. inject into the steam generators
4. inject into the RCS
5. reduce fission product releases
6. control containment conditions
7. reduce containment hydrogen

The guidelines specify a method for a systematic, logical evaluation of the possible strategies and a process of deciding which actions to implement.

Another guideline, SAEG-1, monitors long-term activities after a particular strategy is implemented. Such activities depend on a number of factors, including the equipment put into service to implement the strategies, equipment already in service before implementing the SAMG that relates to the control of a DFC parameter, limitations on equipment usage identified in the guidelines that evaluate the possible strategies, equipment no longer in service if implementation of a strategy is discontinued, and changes in plant conditions following implementation of severe accident management strategies.

A final guideline, SAEG-2 for SAMG termination, comes into play when the plant has been put into a safe, stable state. At this time, selected parameters in the DFC are below their setpoint values and are stable or decreasing, and no new SAMG strategies will be required. However, generic SAMG exit guidance has been developed.

Four computational aids have been developed to assist the TSC staff in diagnosing and formulating appropriate strategies:

1. RCS injection to recover the core
2. injection rate for long-term heat removal
3. hydrogen flammability in containment
4. containment water level and volume

The NRC staff reviewed TR-66, the AP1000 framework document (APP-GW-GL-027), and the executive volume of the AP1000 SAMG. The staff confirmed that the AP1000 SAMG reflects current understanding of severe accident progression. The staff examined the remaining two volumes to ensure that they are an appropriate extension of the EPRI guidance, consistent with the SAMG developed by the WOG for operating reactors, and address all necessary high-level actions. The DCD appropriately references these documents.

Since concerns over hydrogen generation suggest maximizing the flow rate, while concerns about a degraded reactor vessel (overheating and wall thinning at or near the surface of the pool of relocated core material) suggest that flow should be controlled, in RAI-SRP19.0-SPLA-16, the staff asked the applicant to explain how it will provide guidance to resolve potentially conflicting considerations when introducing water to a dry reactor vessel after core relocation (full or partial) to the bottom head.

In a letter dated August 21, 2008, the applicant stated that it will update the AP1000 SAMG for "inject into the RCS" to address the recommended rate of injection into the RCS for situations in which the injection capability is recovered after significant core damage has occurred. The applicant will add a new item in the evaluation of the potential negative impacts of injecting into

the RCS to note that high injection flow rates will minimize the potential for significant hydrogen generation while lower, controlled flow rates will minimize the potential for failure of the reactor vessel. The evaluation will explain that hydrogen generation caused by low flow rates will only be a concern if a significant amount of hydrogen is already in the containment indicating a failure of the hydrogen igniters or the recombiners, or both. On the other hand, the concern about reactor vessel integrity due to high injection flow rates will only exist when a prolonged period has elapsed since the onset of high core temperatures and the reactor vessel will be externally cooled by submergence in water.

Attachment B to the guideline and the associated background document will also provide a full discussion of the conditions under which each of the concerns is applicable. The staff considers RAI-SRP19.0-SPLA-16 resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

On the basis of this review, the staff concludes that SAMG for the AP1000 is consistent with NEI 91-04 and is a logical extension of the WOG SAMG. The discussions of the high-level actions, supported by the background documents provided, establish a sound technical basis for AP1000 COL applicants to develop their severe accident management procedures and training.

For this reason, the staff considers AP1000 COL Information Item 19.59.10-4 to be closed for COL applicants referencing the AP1000 DCD.

19.3 Shutdown Evaluation

19.3.7 Outage Planning and Control

In RAI-SRP19.0-SPLA-10, the staff asked the applicant to clarify whether the development of freeze seal guidelines is the responsibility of the COL applicant; is included in the procedure development described in TR-70, APP-GW-GLR-040, "Plant Operations, Surveillance, and Maintenance Procedures," Revision 1, August 2007; or is controlled in some other way.

The applicant modified DCD Section 13.5 to include the following statement: "If freeze seals are to be used, plant-specific guidelines will be developed to reduce the potential for loss of RCS boundary and inventory when they are in use," and confirmed that COL Information Item 13.5-1 includes the guidelines for use of freeze seals. It is among the guidelines identified as "Phase 3" procedure activities. The staff considers RAI-SRP19.0-SPLA-10 resolved.

NUREG-1793 described other COL action items related to shutdown procedures that the staff consolidated in COL Information Item 13.5-1. In RAI-SRP19.0-SPLA-11, the staff asked the applicant how it would identify these action items to the COL applicant:

- COL Action Item 19.1.8.1-4: The COL applicant will implement the maintenance guidelines as described in WCAP-14837, "AP600 Shutdown Evaluation Report."

- COL Action Item 19.1.8.3-1: The COL applicant is responsible for developing procedures…to close containment hatches and penetrations following an accident during MODE 5 and MODE 6 before steam is released into the containment.

- COL Action Item 19.1.8.16-1: The COL applicant will have policies that maximize the availability of normal residual heat removal (valve V-023) and procedures to open this valve during cold shutdown and refueling operations when the RCS is open and the passive residual heat removal system cannot be used for core cooling.

- COL Action Item 19.1.8.16-2: The COL applicant will develop administrative controls to ensure that inadvertent opening of RNS valve V-024 is unlikely since inadvertent opening results in a draindown of the RCS inventory to the IRWST and requires gravity injection from the IRWST.

- COL Action Item 19.1.8.16-4: The COL applicant will maintain procedures to respond to low hot-leg level alarms.

- COL Action Item 19.3.7-1: The COL applicant will develop an outage planning and control program and will appropriately address the factors that improve low-power and shutdown operations consistent with DCD Tier 2, Chapter 19E, "Shutdown Evaluation," and NUMARC 91-06, "Guidelines for Industry Actions To Assess Shutdown Management."

- COL Action Item 19.1.8.7-1: The COL applicant will implement procedures and policies to have the nonsafety-related wide-range pressurizer level indication during cold shutdown.

Each of these action items has a corresponding entry in DCD Table 19.58-18. COL licensees referencing the AP1000 design must verify that the insights and assumptions documented in this table are satisfied. The staff finds this to be an acceptable method for ensuring that appropriate administrative controls will be applied. The staff considers RAI-SRP19.0-SPLA-11 resolved.

In TR-70, the applicant suggested that the procedures in Phase 3 need not be developed until after a COL is issued.

The staff agrees that these guidelines do not need to be completed at the time of application and finds that the program described in TR-70 is an acceptable method for providing assurance that appropriate guidance will be developed in a timely manner.

19.3.10 Flood Protection

NUREG-1793 states the following:

> The design provides fire detection and suppression capability. The design also provides flooding control features and sump level indication. The COL applicant is expected to take compensatory measures to maintain adequate detection and suppression capability during maintenance activities. This is part of COL Action Item 19.1.8.1-3.

The staff expects the COL licensees to take compensatory measures to maintain adequate detection capability during maintenance activities. The staff identified two COL information items that address fire detection and suppression but not flooding. In RAI-SRP19.0-SPLA-08,

the staff asked the applicant to clarify how it will address concerns about flooding detection, barrier integrity, and control.

The applicant's August 21, 2008, response focused on aspects of the AP1000 design that obviate the need to compensate for a maintenance-related breach of flooding barriers. Specifically, the design does not include any watertight doors; flood barriers are permanent fixtures that are neither opened nor altered by normal activities, including maintenance. Moreover, the CDF contribution from internal flooding is extremely low.

The staff agrees with the applicant's assessment of internal flooding risk. As stated in NUREG-1793, the results of the AP1000 study for internal flooding show that the AP1000 design is adequate because internal floods during shutdown do not represent a significant risk contribution. The staff considers RAI-SRP19.0-SPLA-08 resolved.

19.5 Conclusion

The staff evaluated the information submitted by the applicant in accordance with NUREG-0800 Sections 19.0 and 19.1. In addition to its review of the documents identified above, the staff conducted a regulatory audit on August 9 and 10, 2010.

The applicant has updated the AP1000 PRA to include the most recent I&C design information. Additionally, to facilitate future updates, the applicant converted the PRA software package from the proprietary WesSAGE software package to the CAFTA software package. The applicant also documented the basis for its determination that other design changes did not affect the SSCs modeled in the PRA in a manner that affected the PRA.

In addition to reviewing the description of changes to the PRA, the staff reviewed the description of the new I&C design, specifically the PLS and PMS. The staff reviewed the methods and procedures for conversion of the PRA model from WesSAGE to CAFTA in the applicant's offices. The staff also reviewed the process by which the applicant incorporated the design changes in the PRA.

The staff noted that the applicant has a formal procedure for the review of design packages to ensure that it identifies and addresses any impact on the PRA. Design change packages are evaluated for the potential to alter the PRA model or to affect PRA assumptions or insights.

With the resolution of the open items described in Section 19.1.10 of this report, the staff concludes that the results and insights of the upgraded and updated design-specific PRA for the AP1000 demonstrate that the design meets the Commission's safety goals. These results and insights are an acceptable basis for the risk-informed review of the amended AP1000 DCD.

19F Aircraft Impact

This section describes the staff's evaluation of design features and functional capabilities of the AP1000 that are credited by the applicant to show that the facility can withstand the impact of a large commercial aircraft. These design features and functional capabilities were described in a letter dated March 19, 2010, that proposed changes to AP1000 DCD, Appendix 19F, "Malevolent Aircraft Impact." Upon reviewing the appendix, the staff found that the descriptions of design features and functional capabilities were incomplete. In RAI-SRP19F-AIA-01, the staff identified specific areas in the DCD that required augmentation. In response, the applicant proposed amendments to Appendix 19F and related sections of other DCD chapters.

The impact of a large commercial aircraft is a beyond-design-basis event. 10 CFR 50.150, "Aircraft impact assessment," requires applicants for new nuclear power reactors[7] to perform an assessment of the effects on the designed facility of the impact of a large, commercial aircraft. Applicants for DC are required to submit: (1) a description of the design features and functional capabilities identified as a result of the assessment in their DCD with (2) a description of how the identified design features and functional capabilities meet the acceptance criteria in 10 CFR 50.150(a)(1). Applicants subject to 10 CFR 50.150 must make the complete AIA available for NRC inspection in accordance with 10 CFR 50.70, "Inspections"; 10 CFR 50.71, "Maintenance of records, making of reports"; and Section 161.c of the Atomic Energy Act of 1954, as amended.

Regulatory Criteria

The staff used the following regulations and guidance to perform this review:

Regulations

10 CFR 50.150(a)(1) requires that applicants perform a design-specific assessment of the effects on the facility of the impact of a large, commercial aircraft. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions: (i) The reactor core remains cooled, or the containment remains intact; and (ii) spent fuel cooling or spent fuel pool (SFP) integrity is maintained.

10 CFR 50.150(b) requires that the FSAR include a description of: (1) the design features and functional capabilities that the applicant has identified in accordance with 10 CFR 50.150(a)(1); and (2) how those design features and functional capabilities meet the assessment requirements of 10 CFR 50.150(a)(1).

Review Guidance

DG-1176 provides guidance for meeting the requirements in 10 CFR 50.150(a). It documents NRC endorsement of the methodologies described in the guidance prepared by NEI 07-13, Revision 7.

[7] "Applicants for new nuclear power reactors" is defined in the Statement of Considerations for the Aircraft Impact Rule [74 FR 28112, June 12, 2009].

Supplementary information for the aircraft impact assessment rule [74 FR 28112], which indicates, among other things, that for the NRC to conclude that the rule has been met, it must find that the applicant has performed an AIA reasonably formulated to identify design features and functional capabilities to show, with reduced use of operator action, that the acceptance criteria in 10 CFR 50.150(a)(1) are met.

The following are NRC staff interim review guidelines:

a. Reasonably Formulated Assessment

   The NRC considers an aircraft impact assessment performed by qualified personnel using a method that conforms to the guidance in NEI 07-13, Revision 7, to be a method which is reasonably formulated. The NRC considers qualified personnel to be: (1) an applicant who is the designer of the facility for which the AIA applies; and (2) an applicant's primary contractor for the aircraft impact assessment who has designed a nuclear power reactor facility either already licensed or certified by the NRC or currently under review by the NRC.

b. Reactor Core and Spent Fuel Pool Cooling Design Features

   The "reactor core cooling" criterion or "spent fuel pool cooling" criterion in 10 CFR 50.150(a)(1) is satisfied if design features have been included in the design of the plant to specifically perform that cooling function with reduced use of operator action.

c. Intact Containment

   The "intact containment" criterion in 10 CFR 50.150(a)(1) is satisfied if the containment: (1) will not be perforated by the impact of a large, commercial aircraft; and (2) will maintain ultimate pressure capability, given a core damage event until effective mitigation strategies can be implemented. Effective mitigation strategies are those that provide, for an indefinite period of time, sufficient cooling to the damaged core or containment to limit temperature and pressure challenges below the ultimate pressure capability of the containment as defined in Section 19 of the DCD, Revision 18.

d. Spent Fuel Pool Integrity

   The "spent fuel pool integrity" criterion in 10 CFR 50.150(a)(1) is satisfied if the impact of a large commercial aircraft on the SFP wall or support structures would not result in leakage through the SFP liner below the required minimum water level of the pool.

e. Reduced Operator Action

   The NRC considers use of operator action to be reduced when: (1) all necessary actions to control the nuclear facility can be performed in the control room or at an alternate station containing equipment specifically designed for control purposes; and (2) a reduced amount of active operator intervention, if any, is required to meet the acceptance criteria in 10 CFR 50.150(a)(1). Reduction in the use of operator action is measured relative to the actions required to address aircraft impact without the AIA rule in place (e.g., similar actions in operational programs in place at current operating reactor sites).

19F.1 Summary of Technical Information

Appendix 19F states that the applicant performed an AIA in accordance with the requirements in 10 CFR 50.150(a)(1) using the methodology described in NEI 07-13. Based on the results of the assessment, the applicant has identified a set of key design features to show that the acceptance criteria in 10 CFR 50.150(a)(1) are satisfied. These key design features are reported in Appendix 19F, along with references to other sections of the DCD that provide additional details. Appendix 19F also includes descriptions of how the key design features show that the acceptance criteria in 10 CFR 50.150(a)(1) are met.

19F.1.1 Description of Key Design Features

The key design features that are credited, their function(s), and references to sections including detailed descriptions of the features are summarized below:

Key Design Features

No.	Feature	DCD Reference	Function
1	structural design of the shield building	Chapter 3	protection of safety systems located inside containment
2	structural design of the auxiliary building	Chapter 3	protection of the SFP liner integrity
3	structural design of the wall along the south end of the turbine building at column line 11.2	Section 3.7.2.8.3	protection of the auxiliary building from the impact of a large, commercial aircraft
4	structural design of the wall along the east end of the annex building	Figure 3.7.2-19	protection of the auxiliary building from the impact of a large commercial aircraft
5	structural design and location of the SFP	Figure 3.7.2-12 Section 9.1.2.2	protection of the SFP from the effects of an impact of a large commercial aircraft
6	passive safety injection portions of the PXS and the ADS valves and spargers of the RCS inside containment	Section 5.4 Section 6.3	core cooling

Key Design Features

No.	Feature	DCD Reference	Function
7	physically separated locations of the main control room (MCR), remote shutdown station (RSS), and secondary DAS panel	Chapter 7	manual actuation of passive safety injection and recirculation for long-term core cooling can be initiated if required
8	supporting equipment required for operation of the squib valves from the MCR, RSS, or DAS panel including Class 1E batteries, control and instrumentation cabinets, cabling, and transfer switches	Section 7.7.1.11	manual actuation of passive safety injection
9	ADS Stage 4 squib valves IRWST injection line squib valves and recirculation line squib valves	Section 6.3	passive safety injection and recirculation for long-term core cooling
10	steel containment vessel	Section 19.40	containment integrity maintained with only air cooling for 24 hours
11	normal residual RNS containment isolation valves	Figure 5.4-7	isolation of the RNS outside of the containment
12	reactor trip equipment including sensors, manual inputs, protection and safety monitoring system cabinets, and reactor trip switchgear	Section 7.2.1	reactor shutdown
13	the design and locations of 3-hour fire barriers within the auxiliary building	Section 9.5.1	protection of equipment needed for manual actuation of systems and equipment potentially required for core cooling

Key Design Features			
No.	**Feature**	**DCD Reference**	**Function**
14	specific barriers in the auxiliary building rated to withstand a differential pressure of 34.5 kPa (5 pound(s) per square inch differential (psid))	Section 9.5.1.2.1.1	limitation of the effects of fire damage created by the impact of a large, commercial aircraft

19F.1.2 Description of How Regulatory Acceptance Criteria Are Met

The acceptance criteria in 10 CFR 50.150(a)(1) are that: (1) the reactor core will remain cooled or the containment will remain intact; and (2) SFP cooling or SFP integrity is maintained. The applicant has met 10 CFR 50.150(a)(1) by including features in the AP1000 design that maintain: (1) an intact containment; and (2) SFP integrity following the impact of a large commercial aircraft.

The applicant credits the shield building as a structure that will remain intact following an impact by a large commercial aircraft. Therefore, containment will also be intact. The RPV, PXS, and equipment within the containment will not be damaged by the impact or by exposure to jet fuel. For a postulated impact on the auxiliary building, the applicant credited the design and locations of 3-hour rated fire barriers to limit damages such that manual actuation of core cooling equipment, if necessary, can be achieved.

The AP1000 design also satisfies the SFP integrity acceptance criterion because it is surrounded by barrier walls that protect the SFP liner from adverse effects of an impact by a large commercial aircraft.

19F.2 Evaluation

The staff has reviewed the description of key design features provided by the applicant and the description of how the key design features show that the acceptance criteria in 10 CFR 50.150(a)(1) are met. The staff's evaluation is provided below.

19F.2.1 Reasonably Formulated Assessment

The applicant states that its AIA is based on the guidance of NEI 07-13, Revision 7. In a letter dated August 6, 2010, in response to RAI-SRP19F-AIA-07, the applicant confirmed that the AP1000 assessment does not take exceptions to portions of NEI 07-13 guidance that apply to the AP1000 design. In a revised RAI response dated September 15, 2010, the applicant further stated that an analytical evaluation and experimental verification has been performed for the first-of-a-kind steel-concrete modular design feature subjected to the aircraft impact loading in accordance with the recommendation set forth in Section 2.4.1(4) of NEI 07-13. Based on the applicant's use of NRC-endorsed guidance document, NEI 07-13, Revision 7, by qualified personnel, the staff finds that the applicant has performed a reasonably formulated assessment.

19F.2.2 Key Design Features for Core Cooling

The staff found the initial description of design features and functional capabilities for core cooling to be incomplete. In RAI-SRP19F-AIA-01, the staff identified specific areas in the DCD that needed to be augmented with additional information. The staff's review of the applicant's response to RAIs pertaining to core cooling is discussed below.

In RAI-SRP19F-AIA-01, the staff requested that the applicant identify and describe the specific design features relied upon to maintain core cooling following impact of a large commercial aircraft during power operation. In its response dated September 17, 2010, the applicant identified the following key design features: safety-related passive safety injection and long-term recirculation cooling systems, specific squib valves that need to actuate and equipment that supports actuation of the squib valves, including Class 1E batteries, control and instrumentation cabinets, cabling, and transfer switches. These design features are fully described in DCD Section 6.3 and parts of Section 7.

The staff reviewed the descriptions of these design features and found that the passive safety injection and long-term recirculation cooling system, have been designed specifically to maintain core cooling functions following design-basis events initiated during power operation. The staff considered the descriptions of the features, as well as the ability of these features to perform their design basis safety functions following impact of a large commercial aircraft, including conditions involving loss of coolant from the RCS, and finds that they are suitable for maintaining core cooling following an impact of a large commercial aircraft,. The staff also considered that since the AP1000 AIA credited the shield building to remain intact upon an aircraft impact, the components of the passive safety injection and long-term recirculation systems that are located within the containment structure would not be exposed to jet fuel damage. Furthermore, should this design feature fail to initiate automatically, it can be initiated or operated either from the MCR, the RSS, or the secondary DAS panel, and require no further operator intervention to maintain the core cooling function. This function can be achieved with the key design features identified in the table above as Numbers 1, 6, 8, and 9. On this basis, the staff finds the applicant's description of the key design features for core cooling to be adequate. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In RAI-SRP19F-AIA-02, the staff requested that the applicant identify and describe the specific design features that are relied on to maintain core cooling following impact of a large commercial aircraft while the plant is shut down and the reactor is being cooled by the normal RNS. In its response dated August 6, 2010, the applicant states that should this occur, one of two sets of RNS containment isolation valves located inside containment can be closed from the MCR to terminate leakage from the RCS. These valves and the spatial separation between their location and the location of the MCR are identified as key design features. Core cooling is provided by gravity injection from the IRWST, initially, and the containment recirculation system in the long term. These two systems, including their squib valves, are identified in the DCD as key design features.

AP1000 DCD Section 19.E.2.3.3.1 describes use of these features for core cooling following a loss of RNS during shutdown in MODE 5 with the RCS open. The staff's review of this section of the DCD is described in Chapter 19 of NUREG-1793. The applicant has identified appropriate design features that have been shown to be effective in providing core cooling when the RNS is not available. Based on the above, the staff finds the applicant's description of the key design features for core cooling to be adequate. In a subsequent revision to the

AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

To conform to the guidance in NEI 07-13, applicants must consider whether the impact of large commercial aircraft would cause damage that could prevent the reactor from shutting down in the unlikely event that operators did not manually trip the reactor prior to impact. In RAI-SRP19F-AIA-03, the staff requested that the applicant describe those design features that assure the reactor will be shut down following an aircraft impact, including any features that protect equipment needed for reactor shutdown. In its response dated August 6, 2010, the applicant stated that the equipment needed for reactor shutdown is described in DCD Section 7.2.1 and is considered a key design feature(s). They also indicated that the design of this equipment is such that the reactor trip function will fail in a safe manner (control rods drop into the reactor by the force of gravity) if any of this equipment should become damaged by an aircraft impact. The fail-safe design of this equipment is described in DCD Section 7. Based on the above the staff finds that the applicant has adequately described how the reactor will be shut down should equipment normally relied upon for reactor shutdown be damaged by aircraft impact. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In its initial submittal, the applicant identified the locations of the MCR, RSS and the secondary DAS panel as key design features. In RAI-SRP19F-AIA-04, the staff requested that the applicant describe the roles these features play in satisfying the acceptance criteria in the rule. The staff also requested that the applicant clarify which SSCs in these locations are being credited for actuation of the core cooling equipment. In its response dated August 6, 2010, the applicant stated that manual initiation of core cooling requires operator action from the MCR, RSS, or secondary DAS panel and that each location contains equipment capable of initiating core cooling design features. The applicant also stated that the AIA shows that no single strike from a large commercial aircraft can simultaneously damage the equipment at all three locations. Thus, the degree of separation between these specific facilities is a key feature of the plant design that enables the core cooling acceptance criterion in the rule to be met. The applicant included similar statements in DCD Appendix 19F.4.2. The applicant also added a description of the equipment in these locations needed to actuate core cooling design features. This equipment includes the Class 1E batteries, the supporting PMS control and instrumentation cabinets and cabling for the equipment identified in Appendix 19F.4.3, the transfer switch to isolate the MCR and transfer controls to the remote shutdown room, and the DAS cabling for the squib valve control cabinet.

The staff reviewed detailed descriptions of equipment needed for safe shutdown of the facility in DCD Section 7 and compared them with the set of SSCs cited by the applicant as needed to actuate core cooling design feature. The staff found that the applicant has adequately described key design features needed for actuating core cooling. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19F.2.3 Key Design Features that Protect Core Cooling Design Feature

Structures

In Appendix 19F, the applicant states that the robust shield building design, as described in DCD Chapter 3, is a key design feature that would provide protection for the core cooling key design feature. It also states that the AP1000 assessment concluded that a strike upon the

shield building by a large commercial aircraft would not result in perforation of the shield building and the containment vessel. Because the passive safety injection and long-term recirculation cooling systems that are credited as key design features for core cooling are located inside the containment vessel, it is expected that they will not be damaged, either from an aircraft impact or from exposure to jet fuel.

Fire Protection

The key design features that protect the core cooling key design feature also include the 3-hour rated barriers around and within the auxiliary building as described in DCD Sections 9.5.1 and 9A. The design and locations of fire barriers within the auxiliary building are credited to confine the spread of fire damage resulting from a large commercial aircraft impact. The applicant also credited 5 specific 34.5 kPa (5 psid) fire barriers, as described in DCD Section 9.5.1.2.1.1, to further limit fire spread. The AP1000 assessment determined that no aircraft impact scenario would cause perforation and subsequent fire propagation into the containment where the core cooling equipment is located. Neither would any scenario simultaneously destroy all three redundant locations where support equipment for manual actuation of the core cooling function is located. These key design features are identified in the table above as Numbers 1, 7, 13, and 14. On this basis, the staff finds the applicant's description of the key design features for protecting safety systems required to maintain core cooling to be adequate, as these systems are physically separated and protected by robust structural barriers.

19F.2.4 Containment Intact

The applicant states that the shield building and the protected steel containment vessel as described in DCD Chapter 3 are key design features for maintaining containment intact. Appendix 19F also states that the AP1000 assessment concluded that an aircraft strike upon the shield building would not result in perforation of the shield building and, therefore, the steel containment vessel is not affected. Based on the AP1000 beyond design basis calculation, air-only cooling of containment is sufficient to allow containment integrity to be maintained for 24 hours. This capability is achievable by key design features 1 and 10 as described above. Based on the above, the staff finds the applicant's description of the key design features for maintaining containment intact to be adequate as the containment: (1) will not be perforated by the impact of a large, commercial aircraft; and (2) will maintain ultimate pressure capability.

19F.2.5 Integrity of the Spent Fuel Pool

The key design features credited to maintain the integrity of the SFP are its location and structural design, as described in DCD Section 9.1.2.2 and Figure 3.7.2-12. The applicant indicates that the location and design of the SFP structure are such that it can withstand the effects of an impact of a large, commercial aircraft. The staff finds the applicant's description of the key design features for ensuring SFP integrity to be adequate.

19F.3 Conclusion

The staff finds that the applicant has performed an AIA that is reasonably formulated to identify design features and functional capabilities to show, with reduced use of operator action, that the acceptance criteria in 10 CFR 50.150(a)(1) are met. The staff finds that the applicant adequately describes the key design features and functional capabilities credited to meet 10 CFR 50.150, including descriptions of how the key design features show that the acceptance criteria in 10 CFR 50.150(a)(1) are met. Therefore, the staff finds that the requirements of 10 CFR 50.150(b) are met.

22. REGULATORY TREATMENT OF NON-SAFETY SYSTEMS

22.5.6 Post-72-Hour Actions and Equipment

The staff issued a request for additional information (RAI) on equipment that may be needed 72 hours after an accident (RAI-SRP19.0-SPLA-20). Westinghouse Electric Company, LLC, (the applicant) responded in a letter dated July 15, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091980042), clarifying the description of ancillary equipment for the AP1000.

The AP1000 design relies on the following safety functions 72 hours after an accident:

- core cooling, inventory, and reactivity control
- containment cooling and ultimate heat sink
- main control room habitability
- post-accident monitoring
- spent fuel pool cooling

To support these functions, the AP1000 design includes nonsafety-related ancillary equipment. Connections are provided for generators and pumping equipment that can be brought to the site to back up the installed equipment. The ancillary equipment (alternatively, the transportable equipment) is capable of supporting extended operation of the passive safety systems.

- Electrical power is required to supply the following loads:
  - post-accident monitoring instrumentation
  - spent fuel pool monitoring instrumentation
  - ventilation for the main control room, instrumentation and control room, and direct current equipment room
  - power to replenish the passive containment cooling water storage tank (PCCWST) using motor-driven pumps
- After the energy stored in the 1E batteries is depleted, ancillary diesel generators in the annex building provide alternating current (ac) power. A fuel tank stores sufficient fuel for 4 days of operation (both trains). Power from each ac generator is fed to a distribution panel, from which it supplies the associated 1E battery recharger and passive containment cooling system (PCS) recirculation pump, as well as local services (heating for the ancillary diesel fuel tank and lighting for the ancillary diesel space). All of this equipment is in the seismically qualified portion of the annex building, which is also designed to withstand high winds and associated missiles.
- The seismically qualified portion of the annex building is accessible through the auxiliary building, which is a safety-related, seismic Category I structure. Seven days are available for plant operators to restore at least one path to the outside, through which the

fuel tank can be refilled or a transportable generator can be connected to the distribution panel described above.

- Makeup water is required for the PCCWST, which provides water for the following:
  - containment cooling
  - firefighting
  - spent fuel pool cooling by maintaining the inventory of water in the pool
- The initial inventory of the PCCWST is adequate for 72 hours. An additional 4-day supply of water is stored in the passive containment cooling ancillary water storage tank, which is a seismic Category II structure, and designed to withstand high winds and associated missiles. One of two recirculation pumps in the PCS pumps this water to the PCCWST. If normal power is not available, an associated ancillary diesel generator can power each of the PCS recirculation pumps.
- A nonsafety-related connection is provided for external makeup to the PCCWST. It is compatible with available firefighting equipment and accessible from the yard outside the auxiliary building ("plant west" of the auxiliary building). Through this connection, any additional water required can be injected at the safety-related return line from the PCS recirculating pump and heater to the PCCWST.

The staff concludes that the design features described above provide adequate assurance that required safety functions can be maintained in the long term (beyond 72 hours post-accident) and after seismic events. This is consistent with the Commission's staff requirements memoranda on SECY-94-084 and SECY-95-132, both titled "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994, and May 22, 1995, respectively, and NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." Therefore, the design described above is acceptable.

22.5.9 Short-Term Availability Controls

AP1000 design control document (DCD) Tier 2, Table 16.3-2, "Investment Protection Short-Term Availability Controls," identifies short-term availability controls for nonsafety-related structures, systems, and components (SSCs) that are subject to regulatory treatment.

There are no limiting conditions for operation if the completion times for required actions are not met (i.e., there is no requirement to bring the plant to a safe-shutdown condition when operability requirements are not fulfilled). The staff finds this acceptable since these nonsafety-related systems do not meet any of the four criteria specified in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36(c)(2)(ii), "Technical specifications," that would require a limiting condition for operation:

(1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary

(2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier

(3) an SSC that is part of the primary success path and that functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier

(4) an SSC that operating experience or probabilistic risk assessment has shown to be significant to public health and safety

In addition, inspections, tests, analyses, and acceptance criteria, (ITAAC) as described in DCD Tier 1, Section 3.7, "Design Reliability Assurance Program," address these SSCs. They are described in DCD Tier 2, Section 17.4, "Design Reliability Assurance Program," and identified in Table 17.4-1, "Risk-Significant SSCs Within the Scope of D-RAP"; therefore, the staff finds the administrative controls for regulatory treatment of nonsafety systems in DCD Tier 2, Table 16.3-2, to be acceptable.

In DCD Tier 2, Section 16.3.2, "Combined License Information," the applicant stated that combined license (COL) applicants referencing the AP1000 will develop a procedure to control the operability of investment protection SSCs in accordance with DCD Tier 2, Table 16.3-2. In DCD Tier 2, Section 13.5, "Plant Procedures," the applicant described the commitment to address operational and maintenance programmatic issues. The staff finds that COL Information Item 13.5-1 provides an acceptable method for ensuring that licensees will develop a procedure to control the operability of nonsafety-related SSCs subject to regulatory treatment.

23. DESIGN CHANGES PROPOSED IN ACCORDANCE WITH ISG-11

Westinghouse has submitted information in support of its design certification (DC) amendment application that Westinghouse considers "proprietary" within the meaning of the definition provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390(b)(5). Westinghouse has requested that this information be withheld from public disclosure and the Nuclear Regulatory Commission (NRC) staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff's evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information appears within "square brackets" as follows:

[ ]

The complete text of this chapter, including proprietary information, can be found at Agencywide Documents Access and Management System (ADAMS) Accession Number ML112091879 and can be accessed by those who have specific authorization to access Westinghouse proprietary information.

23. Introduction

This chapter addresses new design changes proposed by Westinghouse (the applicant) that were included in Revision 18 of the AP1000 design control document (DCD). The design changes that are evaluated in this chapter do not constitute all of the changes that the applicant proposed to include in the DCD, Revision 18. Rather, the design changes evaluated in this chapter are in addition to those that the applicant has submitted to the NRC as responses to requests for additional information (RAIs) or safety evaluation report (SER) open items.

The applicant proposes that the DCD be changed by adding new, more detailed design information that expands upon the design information already included in the DCD. This information would be used by every combined license (COL) application that references the certified AP1000 design. The regulations of 10 CFR 52.63(a)(1), "Finality of standard design certifications," require that the Commission may not modify a DC, whether on its own motion or in response to a petition from any person, unless the Commission determines that the change meets one of seven criteria. The staff finds that each of the changes evaluated in this chapter meets one or more of the seven criteria in 10 CFR 52.63(a)(1). Further, the staff finds that many of the design changes evaluated in this chapter contribute to the increased standardization of the certification information because the design amendment would be applied to all COL applicants that reference the DC rule. Therefore, these changes enhance standardization, and meet the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

Interim Staff Guidance, DC/COL-ISG-011, "Interim Staff Guidance Finalizing Licensing-basis Information," describes the staff's position regarding the control of licensing-basis information during and following the initial review of applications for DCs. In part, DC/COL-ISG-11 describes the categories of design changes that applicants should not defer until after the issuance of the DC rule. Categories of those changes that should not be deferred include:

- the correction of significant errors in an application;
- changes needed to ensure compliance with NRC regulations;
- changes needed to support other licensing-basis documents (e.g., conforming changes to information in the final safety analysis report (FSAR) supporting technical specifications (TS));
- significant technical corrections associated with the design or program described in the licensing document (i.e., if not changed, would preclude operation within the bounds of the licensing basis, as opposed to proposed alternatives to the described design or program); and
- changes needed to address a significant vulnerability identified by probabilistic risk assessments (PRAs) or other studies (e.g., a change in a PRA insight).

23.A Changes to Component Cooling Water System

23.A.1 Description of Proposed Changes

In letters dated April 26, 2010, and July 29, 2010, the applicant proposed changes to the design of the component cooling water system. The normal residual heat removal system (RNS), component cooling water system (CCS) relief valves increase in size from 2.54 centimeters (cm) x 2.54 cm (1 inch (in) x 1 in) to 7.62 cm x 10.16 cm (3 in x 4 in) to meet required flow capacity requirements. In addition, the CCS surge tank vent-line increases in size from 5.08 cm x 7.62 cm (2 in to 3 in) nominal pipe size (NPS). Tier 2 text in the DCD is modified to include Sections 3.4.1.2.2.2, "Auxiliary Building Flooding Events"; 9.2.2.3.4, "Component Cooling Water System Valves"; 9.2.2.4.5.2, "Leakage into the Component Cooling Water System from a High Pressure Source"; 19E.2.5, "Component Cooling and Service Water Systems"; and Figure 9.2.2-2, "Component Cooling Water System Piping and Instrumentation Diagram."

23.A.2 Regulatory Basis

The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the reactor coolant system (RCS) is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to regulatory treatment of nonsafety systems (RTNSS) in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related structures, systems, and components (SSCs) or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of inspections, tests, analyses and acceptance criteria (ITAAC), test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for

Nuclear Power Plants," Section 9.2.2, "Reactor Auxiliary Cooling Water System," Revision 4, March 2007, as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified in NUREG-0800 Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

23.A.3 Evaluation

During the staff's evaluation of the proposed CCS design changes, it was determined that additional information was required from the applicant. The staff generated RAI-DCP-CN-06-SBP-01 addressing three issues.

Information previously supplied in a letter dated January 20, 2010, for Section 15.6.2, "Failure of Small Lines Carrying Primary Coolant Outside Containment," was omitted from the April 26, 2010 submittal. This information appeared to be relevant but was removed without an explanation. Additional information is needed to justify that a Chapter 15 change is no longer needed.

The basis for the increased CCS surge tank vent line from 5.1 cm x 7.6 cm (2 in to 3 in) (overflow protection due to normal residual heat removal system leakage into CCS) was not described in DCD Section 9.2.2. The applicant should consider adding this information to the DCD.

The flow rate (as measured in liters per minute (Lpm) (gallons per minute (gpm)) of the CCS/RNS relief valve(s) through the waste water system (WWS) was not described as related to the capacity of the auxiliary building sump pumps (concerns for potential building flooding if the relief valve flow rate exceeds the sump pump flow rate). The applicant should consider adding this information to the DCD.

The applicant's response to RAI-DCP-CN-06-SBP-01 provided the following:

1. The information was deleted from this section in the final proposed change because it describes a non-limiting case (i.e., it is bounded by the sample line break). Discussion of the RNS heat exchanger tube leak has been incorporated into Appendix 19E, Section 19E.2.5.

2. The CCS surge tank line vent/overflow line was increased in size from 5.1 cm x 7.6 cm (2 in to 3 in) to eliminate the potential for over-pressurizing the surge tank (designed as an atmospheric tank) in the event of a large RNS heat exchanger tube leak that causes a significant increase in liquid volume in the CCS. The basis for the increase in surge tank vent line size is described in the proposed revision to DCD Section 9.2.2.3.3.

3. The maximum flow rate possible as a result of a double-ended break of one RNS heat exchanger tube is approximately 1968 Lpm (520 gpm). The large relief valve on the RNS heat exchanger cooling water line discharges to the radioactive waste drain system (WRS) auxiliary building equipment and floor drain sump at elevation 20.3 meters (m) (66 feet (ft)-6 in). This sump is pumped to the waste holdup tank by two air-driven sump pumps, each of which has a nominal capacity of 473 Lpm (125 gpm). In the event that the relief valve discharges continuously for an extended period of time, the WRS floor drain sump may overflow into the 20.3 m (66 ft-6 in) level of the auxiliary building. Section 3.4.1.2.2.2 in Tier 2 of the DCD describes auxiliary building flooding events. In the radiologically controlled area of the first level (elevation 20.3 (66 ft-6 in)) there are no

safe shutdown components and the maximum flood elevation has been determined to be 30.5 cm (12 in) or less, assuming that the flooding is identified and isolated within 30 minutes.

Based on the staff's review, the applicant's response for Item 1 was determined to be acceptable since the DCD Tier 2 text that was deleted for Section 15.6.2 (between the January and the April letters) was not a bounding case and did not need to be included as part of DCD Section 15.6.2. The information that was added to Section 19E.2.5 was evaluated by the staff and was found acceptable since the DCD markup added relevant information associated with the RNS heat exchanger and overpressure protection. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

For Item 2, the staff finds the applicant's response to be acceptable since the increase in relief valve size assures that the CCS surge tank does not over-pressurize due to an RNS heat exchanger tube leak. The existing 5.1 cm (2 in) NPS was too small to pass the approximate 1968 Lpm (520 gpm) leakage from the RNS tube rupture. The DCD markup was provided to add this RNS leak-rate value to DCD Tier 2, Section 9.2.2.3.3. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

For Item 3, the staff finds the applicant's response to be acceptable since an overflow of the WRS floor drain sump does not affect safe shutdown components of a RNS heat exchanger tube break. The maximum flooding level on this floor (auxiliary building, elevation 20.3 (66 ft-6 in) was determined to be 30.5 cm (12 in) or less above the floor, assuming flooding is identified and isolated within 30 minutes.

23.A.4 Conclusion

The staff's review concludes that the design changes described above are acceptable since the proposed changes will not adversely affect safety-related SSCs and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. These design changes were evaluated with respect to the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS and found acceptable. RAI-DCP-CN-06-SBP-01 is resolved.

23.B Changes to Component Cooling Water System

23.B.1 Description of Proposed Change

In letters dated April 26, 2010, and July 29, 2010, the applicant proposed changes to the design of the CCS. Four piping header systems were added for the RNS/CCS relief valves. DCD Tier 2, Figure 9.2.2-2, was modified to include these piping header systems.

23.B.2 Regulatory Basis

The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793. While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing

shutdown risk when the RCS is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to RTNSS in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified by NUREG-0800 Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

23.B.3 Evaluation

During the staff's evaluation of the proposed CCS design changes, it was determined that additional clarification to the description of the design changes was required from the applicant; therefore, the staff generated RAI-DCP-CN-09-SBP-01.

The applicant's response to RAI-DCP-CN-09-SBP-01 is as follows:

> The "Eliminate the RNS ..." statement should be replaced with "Larger pressure relief valves have been added to the CCW system." The reason for this design change is further clarified by:
>
> The original (1" x 1") thermal relief valves provided for the RNS and SFS heat exchangers were intended to discharge to the floor of the CCS valve room, where the discharge would be collected by a nearby floor drain. Now that much higher capacity relief valves are needed for the RNS heat exchanger cooling water lines to meet ASME VIII equipment overpressure protection requirements, the larger valves needed for the RNS heat exchanger (V302A/B) are piped to a dedicated drain collection funnel located in the valve room to minimize the potential release of large quantities of vapor and water spray to the room in the event that a high capacity discharge occurred. The smaller valves (V342A/B) are also provided with their own collection header that discharges near the existing room floor drain. These changes are shown in the revised sheet 3 of Tier 2 DCD Figure 9.2.2-2.

Based on the staff's review, the applicant's response was determined to be acceptable since the design includes a collection piping header and dedicated drain funnel to handle the potential release of large quantities of vapor and water spray to the room in the event of a high capacity relief valve discharge. This discharge connects to the main drain header in the auxiliary building. The previous design allowed relief valve discharge to the floor with water collection to the nearest floor drain. The DCD markup was provided to add this design improvement to DCD Tier 2, Figure 9.2.2-2. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.B.4 Conclusion

The staff' concludes that the design changes described above are acceptable since the proposed changes will not adversely affect safety-related SSCs and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed

changes. These design changes were evaluated with respect to the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS and found acceptable. RAI-DCP-CN-09-SBP-01 is resolved.

23.C Changes to Component Cooling Water System

23.C.1 Description of Proposed Changes

In a letter dated June 18, 2010, the applicant proposed changes to the design of the CCS. The proposed changes increase the size of the relief valves on four cooling water lines associated with the reactor coolant pumps (RCPs), and one chemical and volume control system (CVS). Specifically, the RCP cooling water line relief valves V253/A/B/C/D are changed from 7.6 cm x 10.2 cm (3 in x 4 in) to 10.2 cm x 15.2 cm (4 in x 6 in) with the associated branch lines L253A/B/C/D changed from 7.6 cm (3 in) to 10.2 cm (4 in). Also the CVS letdown cooling water line relief valve V222 is changed from 5.1 cm x 7.6 cm (2 in x 3 in) to 7.6 cm x 10.2 cm (3 in x 4 in) with the associated branch line size of L222 changed from 5.1 cm (2 in) to 7.6 cm (3 in). The basis for the change is to prevent over-pressurization of the CCS piping systems. DCD Tier 2, Figure 9.2.2-2 was modified to include these changes.

23.C.2 Regulatory Basis

The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793. While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the RCS is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to RTNSS in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified in NUREG-0800 Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

23.C.3 Evaluation

Based on the staff's review, these design changes were determined to be acceptable. The changes to the five relief valves and associated branch piping are necessary to prevent over-pressurization of the CCS due to a postulated tube leak in the RCP and CVS heat exchangers with the cooling water lines isolated. The larger relief valves, 10.2 cm x 15.2 cm (4 in x 6 in) for the RCPs and 7.6 cm x 10.2 cm (3 in x 4 in) for CVS, and associated piping have been adequately sized for the required relief flow, thus preventing possible damage to the CCS. The DCD markup was provided to address these changes to DCD Tier 2, Figure 9.2.2-2. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.C.4 Conclusion

The staff's review concludes that the design changes described above are acceptable since the proposed changes will not adversely affect safety-related SSCs and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. These design changes were evaluated with respect to the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS and found acceptable.

23.D Changes to Ancillary Diesel Generator System

23.D.1 Description of Proposed Changes

In letters dated May 10, 2010, July 29, 2010, and August 5, 2010, The applicant proposed changes to the design of the ancillary diesel generator system. These design change apply to the nonsafety-related ancillary diesel generators and include the following changes:

- Increase the rating (from 35 kilowatts (kW) to 80 kW) and physical size of the ancillary diesel generators
- Increase the size of the diesel fuel oil storage tank and change principal construction code to the American Society of Mechanical Engineers (ASME) Code, Section VIII
- Revise certain circuit breaker sizes
- Revise corresponding physical drawings and electrical one line diagrams
- Revise design of the heating, ventilation, and air conditioning (HVAC) system for ancillary diesel generator and oil storage tank rooms

23.D.2 Regulatory Basis

Sections 8.3.1.3 and 9.5.4 of NUREG-1793 address the function and acceptability of the ancillary diesel generators for the AP1000. No regulatory basis documents are identified as applicable to the ancillary diesel generators since they are nonsafety-related, commercial grade equipment. Acceptability of these design changes were based on conformance with the existing AP1000 licensing basis.

23.D.3 Evaluation

The ancillary diesel generators are designed to provide the post-72 hour power requirements following an extended loss of onsite power sources. The ancillary diesel generators are not safety-related but provide electric power for Class 1E post-accident monitoring, main control room (MCR) lighting, MCR and Division B and C instrumentation and controls (I&C) room ventilation and for refilling the passive containment cooling system (PCS) water storage tank and the spent fuel pool (SFP) when no other sources of power are available. The ancillary diesel generators are not required to perform these functions for the first 72 hours following a loss of all other alternating current (ac) power sources. As stated in the proposed design changes submittal, the design of the anchorage for these skid-mounted package units is consistent with the safe-shutdown earthquake (SSE) design of equipment anchorages of

seismic Category II equipment. The off-skid fuel oil piping and fuel oil storage tank are analyzed to show that they withstand an SSE.

During the staff's evaluation of the proposed design changes to the ancillary diesel generator design, it was determined that additional information was required from the applicant to complete the staff's evaluation. The staff generated RAI-DCP-CN-59-SBP-01 requesting clarification of the design changes and justification for the changes to the ventilation system for cooling the diesel. In addition, the description of the design changes did not appear to be consistent with the changes made to the DCD description for this system.

The heat removal arrangement for the operating diesels was changed to include ducting from the discharge of the diesel radiators through a backdraft damper to a hurricane-proof louver on an exterior wall of the diesel enclosure building. There is also a damper and recirculation bypass in the cooling exhaust duct that can be closed in cold weather to recirculate engine heat to the diesel generator room for room heating.

The inlet cooling air is admitted to the diesel generator room by opening a personnel access door on an exterior wall of the enclosure building when the diesel generator is operating. The staff questioned the inconsistency between an exhaust protected against hurricanes and an intake that is not protected during operation of the diesel generator.

The applicant response to RAI-DCP-CN-59-SBP-01 provides sufficient clarification of the design change to support the staff's evaluation and also provides proposed additional changes to the DCD to document the clarifications. With respect to the design for hurricane conditions, the applicant stated that a hurricane is only considered to be an initiating event and is postulated to occur when the ancillary diesel generators are not in operation. When the ancillary diesel generators are not operating, the exterior door used to provide intake of engine cooling air will be closed and this door is designed to withstand a hurricane, including windborne missiles. Since no more than a single initiating event hurricane is postulated for the design basis, hurricane protection is not needed when the ancillary diesel generators are operating and the door that provides inlet cooling air is open.

The change in design code for the ancillary diesel fuel oil storage tank from Underwriters Laboratories (UL) to ASME Code Section VIII provides a more rigorous and extensive set of design, fabrication and testing requirements for the tanks, and is, therefore, acceptable to the staff.

The proposed changes to the ancillary diesel generator do not change the ancillary diesel generators' physical or functional relationship to safety-related SSCs and, therefore, do not increase the potential to adversely affect these SSCs. In addition, the proposed DCD changes include a statement that the ancillary diesel generators and associated SSCs are designed to preclude spatial interaction with any other nonseismic SSC that could adversely interact to prevent the functioning of the post-72 hour SSCs following an SSE. The proposed DCD changes as revised in the response to RAI-DCP-CN-59-SBP-01 are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant proposed to change the size of the ancillary generator from 35 kW to 80 kW because the starting motor current of the PCS recirculation pump, which can draw up to 6.5 times the full load current, was not considered in the generator sizing calculations. In addition, as part of this modification, the applicant proposed to revise the input circuit breaker

size for the regulating transformer from 20 amps to 125 amps, the load test breaker from 100 amps to 60 amps, the PCS pump motor from 20 amps to 50 amps, revise Figure 8.3.1-3 of DCD Tier 2, and update Class 1E 208/120 uninterruptible power supply (UPS) one line diagram (Figure 8.3.2-2) to change "Transportable" AC Generator to "Ancillary" AC Generator. In addition, the applicant proposed other associated physical arrangement drawing changes to accommodate the increased size of the ancillary generator.

The staff reviewed the information provided by the applicant and was concerned that the revised rating of the distribution panel and the size of the output breaker for load testing of the ancillary generator were not compatible to accommodate the higher output current from the upgraded ancillary generator. In addition, the staff found that the proposed change to revise the input circuit breaker size for the regulating transformer from 20 amps to 125 amps did not seem to be adequate. In RAI-SRP 8.3.1-EEB-2, the staff requested the applicant provide its basis for the following:

(1) Utilizing a 125 amp breaker to protect 45 kilovolt amps (kVA) regulating transformer.

(2) Adequacy of the distribution panel rating of 100 amps for the 80 kW ancillary ac generators.

(3) Adequacy of the 60 amps breaker for full load testing of the 80 kW ancillary ac generators.

In a letter dated August 5, 2010, the applicant stated that it had reviewed the adequacy and accuracy of both the rating of the distribution panel and size of each of the breakers on the ancillary diesel generator bus. Based on its review, the applicant revised the distribution panel rating and breaker sizes as follows:

The 125 amp breaker to protect the 45 kVA regulating transformer will be changed from 125 amp breaker to 20 amp breaker, as was originally designed, to a breaker size appropriate to the load on the transformer under ancillary diesel generator operating conditions of 20 amps.

(1) The distribution panel rating will be increased from 100 amps to 225 amps so that it will be protected by the generator output breaker. This generator output breaker will be changed from 100 amps to 150 amps so as to be greater than 125 percent of the generator output.

(2) The 60 amp breaker for full load testing of the 80 kW ancillary generators will be replaced with a 150 amp breaker to allow for full testing of the ancillary diesel generator.

(3) The 50 amp breaker to protect PCS motor will be changed to 100 amps in order to avoid spurious tripping of the breaker on PCS motor start.

The staff has reviewed the above information and concludes that the proposed revised rating of the distribution panel and the size of each of the breakers on the ancillary diesel generator bus are compatible with the revised rating of the ancillary diesel generator. Therefore, the staff finds this concern resolved.

23.D.4 Conclusion

The staff's review concludes that the proposed design changes above are acceptable since the proposed changes will not adversely affect safety-related SSCs and the capability of the ancillary diesel generators to perform their post-72 hour function. These design changes were evaluated with respect to conformance with the existing AP1000 licensing basis and found acceptable. RAI-DCP-CN-59-SBP-01 and RAI-SRP 8.3.1-EEB-2 are resolved.

The staff has reviewed the proposed changes to the ancillary diesel generator design and concludes that the ancillary diesel generator distribution panel and the associated load feeder breakers are sized in accordance with the National Electric Code and, therefore, the proposed changes are acceptable. Also, the staff finds the changes made in Figure 8.3.2-2 to be minor and acceptable.

23.E Changes to Potable Water System

23.E.1 Description of Proposed Changes

In letters dated April 26, 2010, and August 2, 2010, The applicant proposed changes to modify the design of the potable water system (PWS) to add a safety-related loop seal in the PWS piping that penetrates the MCR envelope boundary to prevent in-leakage into the MCR envelope during MCR emergency habitability system (VES) operation.

23.E.2 Regulatory Basis

The applicable regulatory requirement and regulatory guidance are as follows:

- 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 2, "Design Basis for Protection against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake

- Regulatory guide (RG) 1.29, "Seismic Design Classification"

23.E.3 Evaluation

The current PWS is a nonsafety system. The applicant proposed a design modification of the PWS for the consideration of control room envelope integrity under a seismic event. If a pipe break occurs resulting from a seismic event, the current design of the PWS would not maintain the MCR VES pressure boundary. In the proposed design modification, the applicant added an isolation valve, a loop seal, and a vacuum breaker to upgrade portions of the PWS to provide safety-related function for MCR isolation in meeting RG 1.197 for MCR envelope integrity.

To meet the requirements of GDC 2 as it relates to structures and systems being capable of withstanding the effects of natural phenomena including seismic event, acceptance depends on meeting the guidance of the portions of Regulatory Position C.1 of RG 1.29 for the safety-related portions of the system and Regulatory Position C.2 of RG 1.29 for the nonsafety-related portions of the system.

To determine the adequacy of the safety-related function of the system modification, the staff requested more detailed information in RAI-DCP-CN01-SBP-COLP-01. In the RAI, the staff requested the applicant provide additional information pertaining to: (1) a figure and system description for the illustration of the system modification in the DCD; (2) the design basis and maintenance requirements for the loop seal; (3) the DCD changes including both Tier 1 and Tier 2; (4) the single failure discussion for the vacuum breaker; (5) the minimum response time for operator manual actions; (6) flood protection; (7) testing requirements; (8) TS requirements; and (9) MCR envelope integrity.

In a letter dated August 2, 2010, responding to RAI-DCP-CN01-SBP-COLP-01, the applicant provided more detailed information. Based on the review of the additional information, the staff found the following discussion from the RAI responses to form the basis for its finding:

1. Figure 9.2.5-1, "Main Control Room Potable Water System Isolation," and system description are added in the DCD Tier 1 and Tier 2 to illustrate the portions of the PWS that have safety-related function pertaining to preventing in-leakage into the MCR envelope.

2. The loop seal is designed to maintain MCR isolation providing the operator sufficient time to close the PWS isolation valves in the event that the nonseismic PWS piping broke. The potable water tank is located at a higher elevation that maintains water in the loop seal by design. Therefore, no maintenance requirements are needed for the loop seal.

3. DCD changes were made in Tier 1 Section 2.7.1, Table 2.7.1-1 and Table 2.7.1-2, to include the safety-related portions of the PSW. These two tables are referenced by ITAAC Table 2.7.1-4. In Tier 2, changes were made in Table 3.2-3, Table 3.9-12, Table 3.9-16, Table 3.11-1, Section 9.2.5.1.1, Section 9.2.5.4, and Section 14.2.9.1.6, which address the safety design basis, safety-related portions of the PWS in AP1000 Class C, seismic Category I, ASME III-3, inservice test requirements, and environmental qualification.

4. The vacuum breaker is to help ensure that a break in the nonsafety-related potable water supply piping would not cause water to siphon from the loop seal. These pressure/vacuum relief devices are not required to consider single active failures. This is consistent with the implementation of a single vacuum relief device in the automatic depressurization system and VES.

5. The safety-related portions of the potable water seal assure MCR pressure boundary integrity after a design basis event. Since a seismic event is not assumed in the analysis to occur simultaneously with another design basis event (such as loss-of-coolant accident (LOCA)), there are no radiological conditions that would change MCR habitability. VES actuation occurs when there is sustained loss of electrical power. For this situation, the design basis for VES actuation is to provide breathable air for the MCR occupants. The MCR occupants will have a supply of breathable air for 72 hours and the MCR would remain habitable. The operators would be alerted to a loss of air through the loop seal piping by the low differential pressure alarm for the MCR and would close the potable water manual isolation valves. The time for the operator actions is not a critical parameter for the safety-related function.

6. The nonseismic PWS lines in the MCR are limited to the kitchen and restroom areas with line sizes of 2.5 cm (1 in) and smaller. There are no safety-related components that would be adversely affected by a rupture of the nonseismic PWS piping in the MCR kitchen and bathroom areas.

7. The initial test program and inservice testing included the PWS to ensure the integrity of the MCR pressure boundary; as shown in Tier 2 Section 14.2.9.1.6, Table 3.9.6, and Table 3.9-16.

8. TS, Section 3.7.6, "Main Control Room, Habitability System," has been updated to include the safety-related isolation valves of the PWS. The MCR pressure boundary includes the PWS water seal that prevents gas flow through the piping. These TS changes were addressed in the response to RAI-SRP-6.4-SPCV-03.

9. The sanitary drainage system (SDS) is one of the systems that penetrate the MCR boundary. Portions of the SDS, including isolation valves and loop seals, have been modified to safety-related and seismic Category I. The modification of the PWS, DCD Tier 1 and Tier 2 information pertaining to the SDS was revised accordingly. The integrity of the AP1000 MCR boundary was addressed in the response to RAI-SRP-6.4-SPCV-03.

The staff has determined that the proposed modification meets RG 1.29, Position C.2 for the nonsafety-related PSW based on the information in Items (1) and (6) above – that the change portions of the nonsafety-related PWS will be safety-related, and that a failure of the nonsafety-related portions of the system will not adversely affect any of the safety-related functions. Further, the DCD information discussed in Item (3) above demonstrates that RG 1.29, Position C.1 is met for the safety-related portions of the system because appropriate classification designations are specified for the PWS consistent with the approach described in DCD Tier 2, Section 3.2. Based on meeting the guidance of RG 1.29, Positions C.1 and C.2, the staff has determined that the modified PWS meets GDC 2.

The staff reviewed the response to Items (2) and (4) and agrees with the applicant's justifications that: (1) no maintenance is required for the loop seal because the elevation of the water tank is sufficient to provide the water seal without maintenance; and (2) the vacuum breaker is not required to consider single active failures because it is consistent with the VES design for the automatic pressure relieve devices.

The staff reviewed the response to Item (5) for the human factors concern of the minimum response time for operator manual actions. The applicant explained that operator action was not credited for any design basis event. The safety-related (seismic) design of the loop seal and surrounding pipe in coordination with other safety-related systems used to manage design basis events ensure that either the hazardous environment that would necessitate control room isolation is prevented and/or that the control room isolation function is maintained. The manual valves were installed as a defense-in-depth measure to address longer-term evaporation of the water in the loop seal, and for deterministic beyond-design-basis evaluations that simply considered the loop seal to be unavailable. There is no regulation or regulatory guidance applicable to operator manual actions used in this manner.

In reviewing Items (7) through (9), the staff finds that there is sufficient information to address the need for testing the isolation valves and establishing their TS to ensure the integrity of the MCR pressure boundary. The review of the integrity of the MCR pressure boundary is under

NUREG-0800 Section 6.4. The proposed changes were found acceptable in SER Section 6.4.1.3 because the overall effectiveness of the control room envelope is demonstrated through the testing associated with the control room integrity program.

Based on the above, the staff has determined that sufficient information is provided to address the staff's questions posed in RAI-DCP-CN01-SBP-COLP-01 and that sufficient details are provided in the DCD markups to address the safety-related function of the PWS. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.E.4 Conclusion

The staff's review concludes that the design changes described above are acceptable because they meet the requirements and guidance of GDC 2, RG 1.29. RAI-DCP-CN01-SBP-COLP-01 is resolved.

23.F Changes to Reactor Coolant Pressure Boundary Leakage Detection

23.F.1 Description of Proposed Changes

In letters dated January 20, 2010, and July 29, 2010, the applicant proposed changes to revise the licensing basis for unidentified RCS leak detection by removing the N13/F18 containment atmosphere radiation monitor and replacing it with the Flourine-18 (F-18) particulate monitor. The TS Limiting Condition for Operation (LCO) 3.4.9a, and the related discussions in TS Bases B 3.4.7 and B 3.4.9, are revised to reflect the change from the "N13/F18 gaseous monitor" to an "F18 particulate monitor."

In the process of reviewing this proposed design change, the staff identified an operating issue pertaining to RCS leakage detection. Operating experiences at Davis Besse (NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity") indicated that prolonged low-level unidentified reactor coolant leakage inside containment could cause material degradation that could compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary.

23.F.2 Regulatory Basis

The applicable regulatory requirement and regulatory guidance are as follows:

- GDC 30, "Quality of Reactor Coolant Pressure Boundary," as it relates to providing means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage

- 10 CFR 52.79(a)37, "Contents of applications; technical information in final safety analysis report," as it relates to "information necessary to demonstrate how operating experience insights have been incorporated into the plant design"

- RG 1.45, Revision 1, as it relates to "Guidance on Monitoring and Responding to Reactor Coolant System Leakage"

The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the containment atmosphere radioactivity monitoring system design information as described in DCD Sections 5.2.5 and 11.5. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance specified in NUREG-0800 Chapter 16.

23.F.3 Evaluation

GDC 30 requires that means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. RG 1.45 describes acceptable methods for implementing GDC 30.

F18 Particulate Monitor

The staff reviewed the proposed design changes, including changes in DCD Tier 2, Appendix 1A, Sections 3.1.4, 3.6.3.3, 5.2.5.3.3, and 11.5.2.3.1, and TS LCO 3.4.9, Sections B.3.4.7 and 3.4.9. The proposed design changes revise the licensing basis for unidentified RCS leak detection by removing the N13/F18 containment atmosphere radiation monitor and replacing it with an F-18 particulate monitor.

Detector Sensitivity

In the review, the staff determined that the applicant did not provide sufficient information to demonstrate that the newly proposed F-18 particulate radiation monitor (PSS-JE-RE027) sensitivity is capable of detecting the RCS leak rate of 1.9 lpm (0.5 gpm) according to the TS. Therefore, in RAI-SRP11.5-CHPB-05, the staff requested additional information on the analysis demonstrating the sensitivity of the proposed radiation monitor.

Although this design change specifies the radiation monitor sensitivity for particulate radioactivity, the change does not provide an analysis to demonstrate that the specified monitor sensitivity is capable of satisfying the technical basis of using realistic radioactive concentrations in the RCS, as described in RG 1.45, Revision 1.

In a May 14, 2010, response to RAI-SRP11.5-CHPB-05, the applicant provided the analysis demonstrating that the monitor can detect a RCS leak rate [ ] lower than that specified in the AP1000 TS. Based on its review, the staff found that the results of the applicant's analysis were highly and directly dependent upon the F-18 concentration in the reactor coolant and the fraction of F-18 assumed to enter the containment atmosphere.

The applicant derived the coolant concentration from two references [ ] and to be conservative and to minimize the amount of radioactivity leaked from the RCS, the applicant used the lower value in its analysis. The staff's own literature review of F-18 in pressurized-water reactor (PWR) reactor coolant indicated that the concentration could be 0.014 uCi/ml, [ ] which is lower than that used by the applicant.

The applicant estimated the amount of F-18 entering the containment atmosphere to be [ ] a small fraction of the total leaked from the RCS. The applicant derived this fraction from an analysis that estimated the fraction flashed into the space between a RCS coolant pipe and the insulation surrounding the pipe, and the fraction escaping from the insulation into the

containment atmosphere. The applicant estimated the fraction flashed into the space between the pipe and insulation using a method described in NUREG-1320, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," dated May 1988. Using an aerosol transport code, the applicant then calculated the fraction escaping the insulation into the containment atmosphere. The staff independently verified the flash fraction calculation using the method described in NUREG-1320. The staff also concluded that the final escape fraction calculated by the aerosol transport model was conservative because almost all aerosols larger than 1 micron diameter fail to escape into the containment atmosphere, thus greatly reducing the amount of radioactivity reaching the monitor.

Based on its review and independent verification, the staff concludes that the proposed monitor is sufficiently sensitive to detect the TS leak rate. Hence, the staff closed RAI-SRP11.5-CHPB-05.

Response Time

In the letter dated March 12, 2010, the applicant proposed to change the radiation monitor response time for the leak detection from 1.9 Lpm (0.5 gpm) within one hour to 1.9 Lpm (0.5 gpm) within two hours. However, in a May 14, 2010, response to RAI-SRP11.5-CHPB-05, the applicant stated that the radiation particulate monitor is capable of detecting a 1.9 Lpm (0.5 gpm) leak in one hour, and the applicant provided corresponding Tier 2 DCD changes. In a teleconference on July 29, 2010, the applicant clarified that the letter dated May 14, 2010, superseded the early letter regarding the response time. The staff has determined that the response time of detecting 1.9 Lpm (0.5 gpm) leakage in one hour as specified in the May 14, 2010, letter is consistent with the certified design of DCD Revision 15. Based on the clarification and the updated DCD Tier 2 changes, the staff has determined that no change in the response time for the radiation monitor is necessary and, therefore, detecting 1.9 Lpm (0.5 gpm) leakage within one hour is acceptable.

Based on the above, the staff has determined that the proposed F-18 particulate radiation monitor sensitivity and response time pertaining to the RCS detection function are acceptable. Further, using a particulate radiation monitor as one of the leakage detection instruments is consistent with the guidance in RG 1.45, Revision 1. Therefore, the staff has determined that the proposed F-18 particulate radiation monitor is acceptable.

With respect to proposed changes to GTS 3.4.9 and their associated bases, the staff finds these changes, as modified by the applicant's response to RAI-SRP11.5-CHPB-05 acceptable because they reflect the system design and operating information described in DCD Sections 5.2.5 and 11.5.

Davis Besse Operating Experience with RCS Leakage Detection

NRC Bulletin 2002-01 described that the operating experience at Davis Besse in 2002 indicated that prolonged low-level unidentified reactor coolant leakage resulting from nozzle cracking of the control rod drive mechanism inside containment could cause material degradation that could compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary. The question was raised regarding licensees' practices for identifying and resolving degradation of the reactor coolant pressure boundary. Pursuant to 10 CFR 52.79(a)37, relating to the requirement to provide "information necessary to demonstrate how operating experience insights have been incorporated into the plant design," in RAI-DCP-CN45-SBP-01, the applicant was requested to address this issue.

COL Information Item

In a letter dated July 29, 2010, responding to RAI-DCP-CN45-SBP-01, the applicant revised DCD to add a new COL information item, COL Information Item 5.2-3, and a new Section 5.2.6.3 to describe COL Information Item 5.2-3 as follows:

> 5.2.6.3 Response to Unidentified Reactor Coolant System Leakage Inside Containment
>
> The Combined License applicant will provide information to address prolonged low-level unidentified reactor coolant leakage inside containment which could cause material degradation such that it could potentially compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary. This issue could be addressed by operating procedures. The procedures should address operator actions in response to prolonged low level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the Technical Specification (TS) limits to provide operator sufficient time to take actions before the TS limit is reached. The procedures should address identifying, monitoring, trending, and repairing prolonged low-level leakage. The procedures should also define the alarm setpoints and demonstrate that the setpoints are sufficiently low to provide an early warning for operator actions prior to Technical Specification limits. In addition, the procedures should address converting the instrument output to a common leakage rate.

The staff reviewed the RAI-DCP-CN45-SBP-01 response and determined it to be acceptable because the description of the COL information item in DCD Section 5.2.6.3 markups is consistent with the guidance in RG 1.45, Revision 1, pertaining to managing the prolonged low-level reactor coolant system leakage. Therefore, GDC 30 is met based on the conformance to RG 1.45. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.F.4 Conclusion

The staff's review concludes that the design changes described above are acceptable because they meet the requirements and guidance of GDC 30 and RG 1.45, Revision 1. In addition, this design change was evaluated with respect to the adequacy of TS that have been established for the RCS leak detection instrumentation system and found acceptable. COL applications that incorporate by reference the DCD must address the new COL information item described in DCD Section 5.2.6.3. RAI-SRP11.5-CHPB-05 and RAI-DCP-CN45-SBP-01 are resolved.

23.G Changes to Spent Fuel Flood-up Valves Remote Position Indication

23.G.1 Description of Proposed Changes

In letters dated April 26, 2010, and August 3, 2010, the applicant proposed changes to the spent fuel flood-up valves remote position indication. AP1000 DCD, Revision 17 requires several valves that connect to the refueling cavity to have their position status monitored during plant shutdowns to prevent draining of the refueling cavity and the SFP. Tier 1 Table 2.3.7-1 presents a list of these valves. These valves are also designed as seismic Category I

components and identified as such in Tier 1 of the DCD. The applicant has identified in these proposed design changes that the spent fuel pool cooling system (SFS) valves SFS-PL-V031 and SFS-PL-V033 are also required to have their position status monitored during plant shutdowns to prevent draining of the refueling cavity and the SFP, but were not identified as such in the DCD. This design change proposes to modify Tier 1 Table 2.3.7-1 to require that these valves have their position status displayed in the MCR. In addition, a previous design change identified that valve SFS-PL-V075 is required to be locked open during normal operation to provide a flow path during scenarios requiring containment flood-up. During refueling, V075 provides the same function of preventing the draining of the refueling cavity and the SFP as V031 and V033. This design change proposes to modify Tier 1 Table 2.3.7-1 to require valve V075 to have its position indicated in the MCR as well. This design change proposes to add two external Class 1E limit switches (open/closed) to the following isolation valves:

- SFS refueling cavity drain to steam generator system (SGS) compartment isolation valve (SFS-PL-V031)
- SFS refueling cavity drain to compartment sump isolation valve (SFS-PL-V033)
- SFS containment floodup isolation valve (SFS-PL-V075)

23.G.2 Regulatory Basis

The regulatory basis for evaluating these proposed design changes are documented in Section 7.5 and Section 9.1.2 of NUREG-1793. In particular, GDC 61, "Fuel Storage and Handling and Radioactivity Control," requires the fuel storage system to be designed for adequate safety under anticipated operating and accident conditions. The system must be designed with the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions. During refueling operations the reactor cavity is connected to the SFP, and failure of the valves and piping sections identified above could drain the water from the refueling cavity and from the SFP. Acceptability of the proposed design changes was based on conformance with the existing AP1000 licensing basis and the guidance specified in NUREG-0800 Section 9.1.2 (as applicable).

Reviews of the changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h), "Codes and standards," and 10 CFR 52.47, "Contents of applications; technical information." The changes must also conform to the requirements of GDC 13, "Instrumentation and Control," and GDC 19, "Control Room," in 10 CFR Part 50, Appendix A, and should meet guidance in RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.5, "Information Systems Important to Safety." Acceptability was also based on conformance with the existing AP1000 licensing basis and criteria specified in NUREG-0800 Section 7.5 (as applicable).

23.G.3 Evaluation

During the staff's evaluation of these proposed design changes, it was determined that additional information was required from the applicant. The staff identified that the markup of DCD pages affected by the proposed design changes included changes that were not directly related to the justification provided in the design change submittal. In RAI-DCP-CN55-SBP-01.a and RAI-DCP-CN55-SBP-01.b, the applicant was requested to provide justification for these additional changes. In the RAI response letter dated August 3, 2010, the applicant stated that

the changes identified by the staff in RAI-DCP-CN55-SBP-01.a were not part of this proposed design change, but were part of RAI-SRP6.4-SPCV-03, R2. The staff found the applicant's response acceptable because these changes were already evaluated by the staff in Section 6.4 of this report. Therefore, RAI-DCP-CN55-SBP-01.a is considered closed.

In response to RAI-DCP-CN55-SBP-01.b, the applicant stated that the deletions in Tier 2 Table 3.9-16 identified by the staff in RAI-DCP-CN55-SBP-01.b were not deleted from the table, but were moved to the next page. Since this represents an editorial change, the staff finds the applicant's response acceptable and RAI-DCP-CN55-SBP-01.b is considered closed.

In RAI-DCP-CN55-SBP-01.c, the staff identified that the Tier 1 drawings depicting this system did not show all the valves mentioned in the design change proposal (DCP). Therefore, the applicant was requested to update the Tier 1 drawings impacted by this design change. In the RAI response letter dated August 3, 2010, the applicant included a markup of Tier 1 Figure 2.3.7-1, which now included valve SFS-PL-V075. Therefore, the staff finds the applicant's response to be acceptable and RAI-DCP-CN55-SBP-01.c is considered closed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

DCD Tier 2 Section 9.1.3.3.5, "Spent Fuel Pool Cooling System Valves," provides a description of the valve arrangement needed for refueling. It was not clear to the staff if this description was impacted by the proposed change. In RAI-DCP-CN55-SBP-01.d, the applicant was requested to confirm that the configuration description provided in Tier 2 Section 9.1.3.3.5 was still valid and had not been impacted by the proposed change. In the RAI response letter dated August 3, 2010, the applicant stated that the valve configuration description provided in Tier 2 Section 9.1.3.3.5 is still valid and has not been impacted by the proposed change. The staff agrees with the applicant's determination that Tier 2 Section 9.1.3.3.5 is still valid; therefore, RAI-DCP-CN55-SBP-01.d is considered closed.

The staff evaluated Tier 2 Figure 9.1-6 (Sheet 1 of 2) "Spent Fuel Pool Cooling System Piping and Instrumentation Diagram," and identified additional possible refueling cavity drain paths. In RAI-DCP-CN55-SBP-01.e, the applicant was requested to confirm that all refueling cavity drain path isolation boundary piping and components were described in Tier 2 and included in the appropriate Tier 1 sections and tables. In the RAI response letter dated August 3, 2010, the applicant stated that all refueling cavity penetrations and associated isolation valves that are at elevations below the minimum safety level outlined in Tier 2, Chapter 16, TS 3.9.4 are designed as seismic Category I, and have been identified in the appropriate DCD (Tier 1 and Tier 2) section. The applicant also clarified that all refueling cavity penetration lines and associated isolation valves that are not seismic Category I are located at elevations that preclude the possibility of draining the refueling cavity below the minimum safe level for refueling operations. The staff finds that the applicant has properly identified all the possible refueling cavity drain paths in the appropriate DCD sections, and that these drain paths are designed as seismic Category I components. Therefore, RAI-DCP-CN55-SBP-01.e is considered closed.

As documented in NUREG-1793, the staff reviewed and approved the SFS in AP1000 DCD Tier 1 Section 2.3.7 and Tier 2 Section 9.1.3. The staff reviewed the applicant's proposed design changes to the SFS in the AP1000 standard design by using the review procedures described in NUREG-0800 Section 7.5 and requirements of GDC 13 and GDC 19 in 10 CFR Part 50, Appendix A. AP1000 DCD, Revision 17, Tier 1 Table 2.3.7-1 shows that the safety-related display in the MCR is required for the SFS refueling cavity drain to the SGS compartment isolation valve (SFS-PL-V031) and SFS refueling cavity drain to the compartment

sump isolation valve (SFS-PL-V033). These two valves are also required to have their position status monitored during plant shutdowns to prevent draining of the SFP. However, in AP1000 DCD, Revision 17, these two valves were designed without limit switches. To achieve the above required safety-related display functions, two external Class 1E limit switches (open/closed) are provided for valve (SFS-PL-V031) and valve (SFS-PL-V033).

The SFS containment floodup isolation valve (SFS-PL-V075) was added to the SFS by a previous DCP. This valve is required to be locked open to provide a flow path during scenarios requiring containment flood-up. Valve SFS-PL-V075, when closed, also provides the function of preventing the refueling cavity from draining. This function requires the status indication displayed for this valve in the MCR. But, the current valve functional requirements do not provide for remote position indication. Therefore, two external Class 1E limit switches (open/closed) have to be provided for the SFS containment floodup isolation valve to achieve the required safety-related display function.

For the specific DCD changes, the applicant added valve SFS-PL-V075 to DCD Tier 1 ITAAC Table 2.3.7-1 and the new SFS containment flood-up line to DCD Tier 1 ITAAC Table 2.3.7-2. The applicant also added the status of the above three valves to DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirements"; Table 3.11-1, "Environmentally Qualified Electrical and Mechanical Equipment"; Table 3.2-3, "AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment"; Table 7.5-1, "Post Accident Monitoring System"; and Table 7.5-7, "Summary of Type D Variables." The staff finds that the proposed design changes described above are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.G.4 Conclusion

The staff evaluated these proposed design changes against conformance with the existing AP1000 licensing basis and the guidance specified in NUREG-0800 Sections 7.5 and 9.1.2 (as applicable). The staff also evaluated these proposed design changes against the requirements of GDC 61, which requires the fuel storage system to be designed for adequate safety under anticipated operating and accident conditions.

The staff concludes that, based on the description provided in these design changes and the RAI responses discussed above, these proposed design changes are in compliance with the requirements of GDC 61 and follow the guidance provided in NUREG-0800 Sections 9.1.2 and 7.5; therefore, the staff finds the proposed changes to be acceptable.

23.H Changes to the AP1000 Steam Generator Thermal-Hydraulic Data Report

23.H.1 Description of Proposed Changes

In letters dated May 10, 2010; August 4, 2010; and August 12, 2010; the applicant proposed the following changes to the AP1000 steam generator (SG) design:

- In DCD Section 5.4.4.3, the pressure drop through the SG flow restrictor at 100 percent steam flow is changed from approximately 103 kiloPascal (kPa) (15 pounds per square inch (psi)) to approximately 138 kPa (20 psi).

- In DCD Table 5.4-5, the SG design fouling factor is changed from 1.937×10^{-5} to 1.586×10^{-5} m^2-°C/watt (W) (1.1×10^{-4} to 9.0×10^{-5} hr-°F-ft^2/British Thermal Unit (BTU)).

- In DCD Table 10.3.2-2 (and TS Table 3.7.1-2), the main steam safety valve (MSSV) lifting settings and relieving capacities are changed as follows:

Valve Nos.		Lift Setting kPa (psig)		Relieving Capacity 10^6 kg/hr (lb/hr)	
		From	To	From	To
MSSV #1	V030A/B	8170 (1185)	8170 (1185)	0.5940 (1.310)	0.5990 (1.320)
MSSV #2	V031A/B	8246 (1196)	8253 (1197)	0.5987 (1.320)	0.6078 (1.340)
MSSV #3	V032A/B	8329 (1208)	8336 (1209)	0.6105 (1.346)	0.6123 (1.350)
MSSV #4	V033A/B	8405 (1219)	8418 (1221)	0.6151 (1.356)	0.6169 (1.360)
MSSV #5	V034A/B	8487 (1231)	8494 (1232)	0.6205 (1.368)	0.6214 (1.370)
MSSV #6	V035A/B	8563 (1242)	8494 (1232)	0.6214 (1.370)	0.6214 (1.370)

The proposed changes to the MSSV lift settings and relieving capacities are in response to an increased pressure drop calculation for the SG steam outlet nozzle, which impacts the MSSV inlet line losses.

23.H.2 Regulatory Basis

The regulatory basis for evaluating the proposed changes to lift settings and relieving capacity is documented in Chapters 10 and 15 of NUREG-1793. GDC 10, "Reactor Design," and GDC 15, "Reactor Coolant System Design," respectively, in 10 CFR Part 50, Appendix A, require that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that specified acceptable fuel design limits (SAFDLs) and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs). The design basis events in DCD Chapter 15 were analyzed to ensure compliance with GDC 10 and GDC 15.

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the MSSV design and operating information described in DCD Section 10.3. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance in NUREG-0800 Chapter 16.

23.H.3 Evaluation

As discussed in the August 12, 2010, response to RAI-DCP-CN58-SRSB-02, the applicant revised the calculation of the pressure differential across the SG steam outlet nozzle flow restrictor in the thermal-hydraulic data report, which is a document typically provided by the manufacturer to the utility operating the SG. The revised calculation showed an increase of the pressure loss from approximately 103 kPa (15 psi) to 138 kPa (20 psi). Therefore, the applicant proposed to revise DCD Section 5.4.4.3 by stating that the resultant pressure drop through the SG flow restrictor at 100 percent steam flow is approximately 138 kPa (20 psi), rather than approximately 103 kPa (15 psi) in DCD Revision 17. Also, the SG design fouling factor is revised from 1.937×10^{-5} to 1.586×10^{-5} m^2-°C/W (1.1×10^{-4} to 0.9×10^{-5} hr-°F-ft^2/BTU). The applicant stated that this reduction is based on the operating experience of replacement SGs with Alloy 690 tubing since 1989, and SG fouling factor is reduced to offset the increased

pressure loss while still maintaining sufficient margin for the SG heat transfer performance. For the Chapter 15 safety analysis, a higher fouling factor was used where reduced heat transfer to the SG is limiting. The proposed lower value of SG design fouling factor provides increased margin. Therefore, there is no effect on the safety analysis.

Because of the increased pressure loss through the SG flow restrictor, the applicant proposed to change the relief capacities and lift settings of the MSSVs in accordance with ASME Code Section III, Subsection NC-7300. The setpoint of MSSV #6 (V035A/B), which is the highest MSSV setpoint, was lowered to account for the increased pressure losses and still maintain the required relieving capacities. In response to RAI-DCP-CN58-SRSB-01, the applicant stated that for the purposes of the AP1000 safety analysis, the significant MSSV setpoints are MSSV #1 (V030A/B) and MSSV #6 (V035A/B). MSSV #1 represents the lowest safety valve setpoint and is used to determine if design transients will challenge the MSSVs. MSSV #6 is the highest valve setpoint and is used to determine the overall steam pressure relief capacity in the safety analysis. In this proposed change, the lift setpoint of MSSV #6 is reduced from 8563 kPa (1242 psig) to 8494 kPa (1232 psig) to account for the impact of the revised SG flow restrictor pressure loss, but the relieving capacity of MSSV remains unchanged. The reduced lift setting resulted in a marginal increase in valve capacity with respect to the safety analysis of the limiting overpressure turbine trip event. Also, the lift setting of MSSV #1 is not changed while its relieving capacity is increased slightly. The setpoints and relieving capacities of MSSV #2 through MSSV #5 are also slightly increased, but they do not affect the safety analyses. Therefore, the proposed slight changes to the MSSV setpoints and relieving capacities have minimal effect on the safety analyses of the overpressure events. For the limiting turbine trip event, the minimum departure from nucleate boiling ratio (DNBR) remains well above the safety analysis DNBR limit, and the RCS pressure and the secondary side steam pressure remain within 110 percent of the respective design pressures.

With respect to proposed changes to TS 3.7.1, the staff finds these changes acceptable because they reflect the MSSV design and operating information described in DCD Section 10.3. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text.

23.H.4 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes are acceptable because these changes have minimal effect on the safety analysis, and GDC 10 and GDC 15 continue to be complied with. These design changes were also evaluated with respect to the adequacy of TS that have been established for the MSSVs. The staff finds these changes acceptable.

23.I Changes Related to the Implementation of P-17 for Rod Withdrawal Prohibit

23.I.1 Description of Proposed Changes

In a letter dated May 10, 2010, the applicant proposed design changes related to the implementation of P-17 logic for rod withdrawal prohibit. The current design requires the P-17 signal coincident with the Beacon Unavailable Signal to generate the automatic rod withdrawal prohibit. The proposed changes would remove the Beacon Unavailable Signal and the associated AND logic gate to enable an automatic rod withdrawal prohibit solely on the rate of change in nuclear power (P-17). The implementation of the P-17 logic to prohibit rod withdrawal is a conservative change from the current design. As a result of this proposed P-17 logic

change, the applicant proposed to revise DCD Section 15.4.3.2 by revising the sequence of dropped rod event in DCD Table 15.4-1 and DCD Figures 15.4-1 through 15.4-4 to reflect this change.

23.I.2 Regulatory Basis

The regulatory basis for evaluating the changes related to the implementation of P-17 logic for rod withdrawal prohibit is documented in Chapter 15 of NUREG-1793. The proposed design changes must also conform to the requirements of GDC 10, GDC 13, GDC 20, "Protection System Functions," and GDC 25, "Protection System Requirements for Reactivity Control Malfunctions" in 10 CFR Part 50, Appendix A.

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed such that SAFDLs are not exceeded during normal operation, including the effects of AOOs. Control rod withdrawal is an AOO. The fuel cladding is the first barrier of protection against radioactive release. Meeting GDC 10 ensures that the fuel cladding integrity is not challenged during this AOO.

GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to ensure adequate safety, and the provision of controls that can maintain these variables and systems within prescribed operating ranges. Meeting GDC 13 ensures that the appropriate controls are provided to maintain these variables and systems within the prescribed operating ranges.

GDC 20 requires that the protective system automatically initiate the operation of the reactivity control system to ensure that fuel design limits are not exceeded as a result of AOOs. The withdrawal of a control assembly significantly impacts local fuel pin power and could lead to cladding failure. Measures are required to ensure that an abnormal rod withdrawal is detected and automatically terminated before fuel design safety limits are violated. Meeting GDC 20 ensures that cladding integrity is not challenged during this AOO.

GDC 25 requires that the reactor protection system be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods. A failure of the reactivity control system that would create an unmitigated withdrawal of a control assembly could lead to cladding failure. Meeting GDC 25 ensures that a power transient fostered from a reactivity addition as a result of a single failure of the reactivity control system will be detected and terminated before challenging the fuel cladding integrity.

23.I.3 Evaluation

The proposed changes to enable the P-17 automatic rod withdrawal prohibit solely on the rate of change in nuclear power affect the response to dropped rod control cluster assembly (RCCA) events, described in DCD Section 15.4.3.2.1. A drop of one or more RCCAs from the same group results in an initial reduction in the core power and a perturbation in the power distribution. In the automatic control mode, the plant control system detects the drop in the core power and initiates withdrawal of a control bank, which could result in power overshoot. The implementation of the proposed P-17 logic to prohibit automatic rod withdrawal prevents the potential power overshoot and is, therefore, conservative relative to the current design. Therefore, the safety analysis of dropped RCCA events would be bounded by the existing

analysis described in DCD Section 15.4.3.2.1, which demonstrated that the minimum DNBR for one or multiple rod drops from the same group is greater than the DNBR limit.

The applicant has run a spectrum of transients, varying key input parameters to verify that DNBR limits continue to be met with the proposed P-17 logic implementation. The sequence of a rod drop event displayed in DCD Table 15.4-1 is considered to be representative of all of the cases run. The results of a representative dropped RCCA event are provided in revised Figures 15.4.3-1 through 15.4.3-4. DCD Table 15.4-1 is also modified to reflect a representative case where the peak nuclear power occurs at time 21.7 seconds and peak core heat flux occurs at 24.2 seconds.

The staff has reviewed the proposed design changes to implement the P-17 logic for rod withdrawal prohibit. Based on its review, the staff finds that this change does not affect the safety analysis of the dropped RCCA events described in DCD Section 15.4.3.2.1, and the analysis continues to satisfy the acceptance criteria of NUREG-0800 Section 15.4.3 with respect to the minimum DNBR, peak pressure, and fuel cladding integrity. Therefore, the staff concludes that the proposed design changes are acceptable.

23.I.4 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 proposed design changes are acceptable because they meet the requirements of GDC 10, GDC 13, GDC 20, and GDC 25.

23.J Changes Related to Post-Design Basis Accident Transmitters

23.J.1 Description of Proposed Changes

In letters dated May 25, 2010, July 29, 2010, and October 20, 2010, the applicant proposed design changes to relocate seven containment pressure transmitters outside containment and connect to remote pressure sensors inside containment by sealed capillary tubing. These changes require the addition of four new containment penetrations, one for each safety division. The applicant also proposed to relocate 18 Category 1 post accident monitoring system (PAMS) transmitters above the maximum design-basis accident (DBA) flood level. In addition, the applicant proposed to reduce post-accident operability time for 18 Category 2 PAMS transmitters from 4 months to 2 weeks.

23.J.2 Regulatory Basis

The regulatory basis for evaluating the changes to post-DBA transmitter requirements is documented in Chapters 3, 5, 6, 7, and 9 of NUREG-1793.

Reviews of the changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of GDC 13 in 10 CFR Part 50, Appendix A, and should meet the guidance in RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment"; RG 1.97, and RG 1.151, "Instrument Sensing Lines." The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.5, "Information Systems Important to Safety." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified by NUREG-0800 Section 7.5 and RG 1.11, RG 1.97, and RG 1.151 (as applicable). The addition of four new containment penetrations was evaluated using the guidance provided in NUREG-0800

Section 6.2.4, "Containment Isolation System"; NUREG-0800 Section 6.2.6, "Containment Leakage Testing"; and the requirements of 10 CFR Part 50, Appendix J, "Containment Leak Rate Testing," and GDC 52, "Capability for Containment Leak Rate Testing."

23.J.3 Evaluation

As documented in NUREG-1793, the staff reviewed and approved the post-DBA transmitter requirements in AP1000 DCD, Revision 15. The staff reviewed the applicant's proposed design changes using the review procedures described in NUREG-0800 Sections 6.2.4, 6.2.6, and 7.5, requirements of GDC 13 and GDC 52 in 10 CFR Part 50, Appendix A; 10 CFR Part 50, Appendix J; and the guidance in RG 1.11, RG 1.97, and RG 1.151.

The applicant proposed to relocate seven containment pressure transmitters outside containment. These changes will allow direct measurement of differential pressure across the containment shell. These changes also allow those seven transmitters to be located in a mild environment. Relocation of the seven transmitters outside of containment could address the LCO in TS that containment pressure shall be maintained between -1.38 kPa (-0.2 psig) to 6.9 kPa (+1.0 psig), which are used in safety analysis.
[

] Relocation of the seven transmitters outside containment allows the plant to operate within TS for containment pressure. Moving the transmitters outside containment also eliminates the need to include a DBA environmental allowance in the determination of channel accuracy and the setpoint for PCS actuation.

Moving seven containment pressure transmitters outside containment and connecting to remote pressure sensors inside containment by sealed capillary tubing requires the addition of four new containment penetrations, one for each safety division. In Divisions A, B and C, the normal-range and wide-range transmitters share one capillary line and penetration. For Division D there is no wide-range transmitter, so the normal-range transmitter is on its own capillary. These four new instrument penetrations, P46, P47, P48, and P46 are identified in marked up DCD Tier 1 Table 2.2.1-1 and Figure 2.2.2-1, "Containment System." In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," or GDC 56, "Primary Containment Isolation," usually require each line that penetrates the containment and is either a part of the reactor coolant pressure boundary or that connects directly to the containment atmosphere, to be provided with containment isolation valves. However, NUREG-0800 Section 6.2.4 endorses the containment isolation provisions described in RG 1.11 for instrument sensing lines. RG 1.11 finds sensing lines with no isolation valves acceptable as long as the lines are sized to limit the potential offsite exposure to be below the guidelines of 10 CFR Part 100, "Reactor site criteria." The capillary tubing meets this criterion.

The four new containment penetrations to accommodate the new instrument capillary tubing will be leak rate tested, as required by GDC 52. The four new penetrations will be Type A tested, which meets the requirements of 10 CFR Part 50, Appendix J and the guidance of NUREG-0800 Section 6.2.6. This leak rate testing commitment will be added to DCD Tier 2, Table 6.2.3-1, "Containment Penetrations and Isolation Valves." In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant also proposed to move 18 Category 1 PAMS transmitters above the maximum DBA flood level within containment. This change is made to ensure that the 18 Category 1 PAMS transmitters are available following the DBA flood. [

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The applicant also proposed to change the post-accident operation period for instruments that monitor Category 2 parameters. These changes would not affect Type A, B, or C primary post-accident parameters. These proposed design changes reduce the post-accident operation period from 4 months to 2 weeks. The Category 2 PAMS transmitter instruments are not required long term following a DBA, so their post-accident operation period can be changed from 4 months to 2 weeks. As Category 2 parameters, these parameters are not considered to be primary post-accident parameters for Type A, B, or C and, therefore, are not required to be qualified long-term following a DBA.

In the markup for AP1000 DCD, Tier 1, Table 2.2.2-1, the applicant changed the qualification for harsh environment from "Yes" to "No" for the seven containment pressure sensors. This change is not justified because the seven containment pressure sensors are still inside containment, even though the seven transmitters are moved outside containment. The applicant also failed to identify the 18 Category 1 PAMS transmitters that are proposed to be relocated above the maximum DBA flood level. As a result, the staff issued RAI-DCP-CN64-ICE-01 requesting the applicant to justify the changes in the environmental qualification of the seven containment pressure sensors. Also, the applicant was requested to identify which Category 1 PAMS transmitters are relocated.

In the RAI response, the applicant restored a harsh environment qualification requirement for the seven containment pressure sensors that are still located inside containment. In the response, the applicant also identified all 18 Category 1 PAMS transmitters that are relocated above the maximum DBA flood level. After reviewing all proposed design changes and information provided in the RAI response, the staff finds that the design changes are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

These proposed changes also affected the TS Bases for Engineered Safety Feature Actuation System Instrumentation, B 3.3.2. The staff found this change acceptable since it provides details of the system design described in the changes to Section 6.2.2 of the DCD.

23.J.4 Conclusion

After reviewing the proposed design changes, markups to DCD Tier 1 and Tier 2 and RAI responses, the staff finds that the proposed design changes meet the post-DBA monitoring requirements for the transmitters. Therefore, the staff concludes that the proposed design changes are acceptable.

23.K Changes to Startup Feedwater System and Chemical and Volume Control System Isolation Logic

23.K.1 Description of Proposed Changes

In letters dated May 10, 2010, and July 29, 2010, the applicant proposed design changes to add an AND logic to the protection and safety monitoring system (PMS) Functional Diagram Sheets 6 and 10 of Figure 7.2-1 in the AP1000 DCD to isolate the startup feedwater system (SFW) and close the CVS isolation valves earlier in the steam generator tube rupture (SGTR) transient sequence in order to maintain the margin to the SG overfill. The SG narrow range level high coincident with reactor trip limiting setpoint is proposed to be changed to 85 percent and the actuation signal added to Table 7.3-1, Table 15.0-6, and TS Table 3.3.2-1 and its Bases. The newly added AND logic is to combine SGS narrow range level high with P-4 reactor trip.

23.K.2 Regulatory Basis

The regulatory basis for evaluating the proposed changes of SFW and CVS isolation on SGS narrow range level high coincident with P-4 reactor trip is documented in Chapter 7 of NUREG-1793.

Reviews of the changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of GDC 13 and GDC 20 in 10 CFR Part 50, Appendix A. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.3, "Engineered Safety Features Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified by NUREG-0800 Section 7.3 (as applicable).

23.K.3 Evaluation

As documented in NUREG-1793, the staff reviewed and approved the engineered safety features in Section 7.3 of AP1000 DCD Tier 2. The staff reviewed the applicant's proposed design changes to the SFW and CVS isolation logic by using the review procedures described in NUREG-0800 Section 7.3 and requirements of GDC 13 and GDC 20 in 10 CFR Part 50, Appendix A.

The containment pressurization backpressure was not considered in the original transient analysis for the DBA of SGTR. However, the increase of the containment pressure resulting from heat transfer to containment via the passive residual heat removal (PRHR) heat exchanger (HX) impacts the boiling temperature of the in-containment refueling water storage tank (IRWST) water and results in a longer duration of SGTR breakflow. The analysis also showed that the SG overfill occurs at 12,739 seconds during a design basis SGTR accident with no operator actions modeled. These design changes have been proposed in order to maintain SG overfill margin without reliance on operator actions.

For the specific DCD changes, the applicant mainly added an AND logic to the PMS software logic (Sheets 6 and 10 of Figure 7.2-1), which combines the existing SG narrow range level high signal from either of the two SGs and P-4 reactor trip signal to isolate SFW and close CVS isolation valves. The narrow range level high signal for each SG results from a coincidence logic of two out of four divisions with bypass functionality. This change meets the reliability and testability requirements in GDC 21, "Protection System Reliability and Testability."

Sections 7.3.1.2.13 and 7.3.1.2.15 in AP1000 DCD Tier 2 are also revised to include the new isolation signal for the SFW and CVS systems. The SG narrow range level high limiting setpoint at 85 percent and the actuation signal are added to Table 7.3-1, Table 15.0-6 and TS Table 3.3.2-1 and its bases. The staff finds that these design changes are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made changes to the DCD text which are consistent with the proposed changes.

The AP1000 design provides automatic protection actions to mitigate the consequences of a SGTR event. The automatic actions include reactor trip, actuation of the PRHR HX, initiation of core makeup tank flow, termination of pressurizer heater operation, and isolation of CVS and SFW on high SG narrow range water level. The proposed change to add the SG narrow level high coincident with P-4 reactor trip would isolate the CVC and SFW at a relatively lower SG level and is, therefore, conservative during the SGTR. This added trip isolation function would conservatively maintain the SG overfill margin without relying on operator actions.

The staff has reviewed the proposed change to add an AND logic to the PMS software that combines the existing SG narrow range level high alarm and P-4 reactor trip to isolate SFW and CVS. Based on its review, the staff finds that the results of the analysis continue to show that the SG will not overfill with water, the maximum RCS pressure will not exceed 110 percent of design pressure, and the minimum DNBR will remain greater than the safety DNBR limit. Therefore, the staff concludes that this proposed change is acceptable.

The proposed design changes also include the addition of isolation functions to the engineered safety feature actuation system (ESFAS). If a narrow range high SG level signal is received, coincident with a P-4 reactor trip, signals are sent to isolate the CVS and SFW. In the case of CVS, this prevents additional make-up to the RCS in case of a SGTR casualty. The SFW isolation prevents an overfill condition and, therefore, an over cooling condition, if the reactor is tripped and the SFW system is not isolated. These changes affect the TS for ESFAS instrumentation, Table 3.3.2-1, as well as the discussion in bases Sections B 3.3.2 and B 3.4.17. The staff finds these changes acceptable since they conform to the guidance provided in the Standard Technical Specifications (STS) and add appropriate isolation for protection in these events, as discussed in the changes to Section 15.6.3 of the DCD.

23.K.4 Conclusion

After reviewing the proposed design changes and markups to DCD Tier 2 sections and tables, the staff finds that the proposed changes meet the applicable requirements and guidance for the safety-related functions. Therefore, the staff concludes that these design changes are acceptable.

Based on the above evaluation, the staff concludes that the proposed design changes are acceptable because the SGTR analysis continues to show margin to SG overfill, and meet the pressure and core safety DNBR limits, even though they are not required to be met for a SGTR event.

23.L Changes to Passive Core Cooling System Injection Lines

23.L.1 Introduction

In letters dated May 25, 2010; August 2, 2010; and August 20, 2010, the applicant proposed design changes to the passive core cooling system (PXS) injection lines to address gas

intrusion, which included the addition of manual vent valves, pipe stubs, manual drain valves, instrumentation, and re-routing accumulator discharge line connections to the direct vessel injection (DVI) lines. These manual vent valves are located in containment rooms that are constructed to permit entry during full power conditions. Analyses have shown that with enough noncondensible gas accumulation, IRWST injection through the affected flow path could be delayed. Therefore, excessive amounts of noncondensible gas accumulation in the high point vents of the IRWST injection lines may potentially impact the passive injection of IRWST borated water into the reactor vessel (RV). However, the presence of a small amount of noncondensible gases does not imply that the IRWST injection capability is immediately inoperable but rather that gases that are accumulating need to be vented. The venting of these gases requires containment entry to manually operate the vent valves. Since gas accumulation is a slow process, plant operators have sufficient time to vent the noncondensible gases upon receiving an alarm. To incorporate the proposed change, the system modification would include revisions to Tier 1 Figure 2.2.3-1, Tier 2 Figures 5.1-5 and 6.3-1, and Tier 2 Tables 3.2-3, 3.9-17, 3.11-1 and 3I.6-3 to identify the new components. In addition, the proposed change includes three new DCD Sections 6.3.6.3, 6.3.6.3.1, and 6.3.6.3.2, which provide a description of the plan to mitigate gas intrusion and accumulation by means of periodic system surveillance and venting procedures, review of pipe layout and routing drawings to identify high-point vent and low-point drain locations, and assessment of system design features. Also, to ensure proper operational implementation of this design change since gas accumulation has the potential to impact safety-related systems, IRWST required actions and surveillance requirements are added to DCD Section 16.1, "Technical Specifications." For controls of IRWST operations, the affected TS sections are TS 3.5.6, TS 3.5.7, and TS 3.5.8 for during Modes 1 through 4, Mode 5, and Mode 6, respectively.

23.L.2 Regulatory Basis

GDC 27, "Combined Reactivity Control Systems Capability" in 10 CFR Part 50, Appendix A, requires that the emergency core cooling system (ECCS) be designed to provide the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

GDC 27 is applicable because upon actuation the ECCS provides rapid injection of borated water to ensure reactor shutdown. Injection of borated water provides negative reactivity to reduce reactor power to residual levels and ensures sufficient cooling flow to the core. Meeting the requirements of GDC 27 for the ECCS augments the protection for the primary fission product barrier by providing a means to ensure that the core, under postulated accident conditions, can be safely shut down and will be maintained in a coolable geometry. Noncondensible gas accumulation has the potential to delay injection of borated water. Such a delay would impact the moderating and heat removal capabilities, thus providing an adverse challenge to the primary fission product barrier and maintenance of coolable core geometry.

GDC 35, "Emergency Core Cooling," requires, among other things, that the ECCS be designed to provide an abundance of core cooling to transfer heat from the core at a rate such that fuel and clad physical damage that could interfere with continued effective core cooling is prevented.

GDC 35 is applicable because following a breach in the reactor coolant pressure boundary; reactor coolant is lost at a rate determined by several factors, including break size and RCS pressure. The ECCSs are relied upon to inject adequate cooling water into the RCS during a LOCA and to circulate the water through the core to provide for core cooling. The ECCS must inject cooling water at a rate sufficient to ensure that the calculated changes in core geometry

will be such that the core remains amenable to cooling, and that the calculated cladding oxidation and hydrogen generation meet the specified performance criteria. Based on analysis, noncondensible gas accumulation has the potential to delay injection of borated water; such a delay may adversely affect fuel and cladding physical configuration with potential to challenge the coolability of the core geometry.

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the IRWST design and operating information described in DCD Section 6.3. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance specified in NUREG-0800 Chapter 16.

23.L.3 Evaluation

Noncondensible gas accumulation effects on the performance of safety related "active" emergency core cooling, residual heat removal, and containment spray systems have been documented based on operating plant experience and system analysis. Some of these events are identified in NRC generic letter (GL) 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems," which illustrates the need for license holders to review their safety systems to resolve this issue. The focus of this GL is centered on four principal concerns: (1) licensing basis, (2) design, (3) testing, and (4) corrective actions. Also, the GL provides an assortment of pertinent generic communications documents related to this subject, which was helpful in the staff's evaluation. Although the GL refers to active safety systems focusing on gas accumulation effects on pump performance, the applicant expanded its existing design activity to include AP1000 passive safety systems to "integrate the draft Interim Staff Guidance (ISG) document ISG-019 regarding gas intrusion assessment guidance into the design process, helping to confirm that the potential issues identified in" the GL have been addressed in the AP1000 design. However, this safety evaluation is related to proposed design changes that primarily address gas intrusion and accumulation in the PXS.

The proposed changes include the addition of the following components:

- 8 manual maintenance vent valves in 6 PXS passive injection and recirculation line piping high point locations;

- 4 pipe stubs with maintenance vents and associated valves, line routing to tee into core makeup tank (CMT) vent line routing to the reactor coolant drain tank (RCDT);

- Remote pipe stub gas void indications at the outlets of each of the IRWST passive injection squib valves;

- Re-routing the 2 accumulator discharge line connections to both DVI line vertical cold trap (riser) pipes to tee in physically above (in elevation) and downstream of the associated IRWST connection into the DVI riser pipes (instead of locating the accumulator connection upstream of and below the IRWST tee);

- 20 manual maintenance drain valves in 14 PXS passive injection and recirculation piping locations, 5 normal RNS piping locations, and 1 RCS piping location. (Note: The RNS

and RCS drains are unrelated to gas intrusion effects but part of design finalization to consolidate required PXS line changes and piping analyses.)

Issues identified during staff review were addressed in three RAIs, identified as RAI-DCP-CN66-SRSB-01 through RAI-DCP-CN66-SRSB-03, with the primary issues summarized below:

1. RAI-DCP-CN66-SRSB-01
   - Explain the exclusion of vent valves to Train B of the containment recirculation.

2. RAI-DCP-CN66-SRSB-02
   - Provide the bases for selecting 5.7 Liter (L) (0.2 cubic feet ($ft^{3)}$) as the noncondensible gas volume limit in the TS surveillance requirements (SR).
   - Describe the volume measuring method of the noncondensible gas.
   - Describe how the 5.7 L (0.2 ft^3) volume of noncondensible gas is accounted for in the safety analyses for LOCA and post-LOCA long term cooling.

3. RAI-DCP-CN66-SRSB-03
   - Provide justification for excluding the new instrumentation and valve components in ITAAC.
   - Describe the calibration frequency of resistance temperature detector (RTD) switches used to measure the volume.
   - Discuss whether the calibration frequency is controlled by TS.

In its August 20, 2010 response to RAI-DCP-CN66-SRSB-01, the applicant stated that containment recirculation Train A and Train B paths have different configurations to avoid other components and piping lines. Whereas, Train A was routed with two high points to circumvent interference with other plant components, Train B layout constraints were less complicated; hence, there are no local high points that could accumulate gas. After further review, the staff finds the RAI response acceptable.

With regard to RAI-DCP-CN66-SRSB-02, the applicant responded that originally the 5.7 L (0.2 ft^3) noncondensible gas volume limit was selected for the CMT inlet high point pipe TS SR 3.5.2.4 because it was slightly larger than the internal volume between the sensors location on the pipe stubs and the first normally closed manual vent valve that was previously evaluated and approved. Since the gas intrusion and accumulation is a slow process and the alarm occurs before voiding is extended into the line, the relocation allows sufficient time for the operators to vent the line before gas buildup could adversely affect the passive safety operability performance. Therefore, initially the PXS volume limit of 5.7 L (0.2 ft^3) was selected to be consistent with the CMT SR requirements since the IRWST injection line high points, pipe stubs, and sensor configuration is similar to that of the CMT. However, based on the applicant's response, the staff concludes that the sensor location not the volume limit is relevant to proper performance of venting the line because the sensor is configured as a level switch, as discussed below. Therefore, the applicant has proposed to remove the reference to the volume limit from TS SRs 3.5.2.4, 3.5.4.3, and 3.5.6.3 and replace it with "has not caused the high-point-water level to drop below the sensor." Also, the corresponding TS Bases would be

revised to appropriately reflect the removal of the volume limit. The staff finds this proposed change acceptable because it refers to the correct function of the sensor.

In addition, the applicant stated that the volume is not directly measured but the change of state from liquid to gas provides the mechanism that controls the sensor output. The sensors are thermal dispersion sensors consisting of one heated RTD and one non-heated RTD configured such that the RTDs function as a thermal dispersion level switch where the temperature difference is based on the conductivity of the medium in contact with the two elements. When the RTDs are exposed to gas, the change in the differential temperature of the elements causes the output switch to actuate providing an alarm to the operator. This sensor configuration has been used in other plant applications with reliable results. Therefore, the staff finds this method to be acceptable.

With respect to safety analyses for LOCA and post-LOCA long term cooling, the applicant stated that the proposed volume is not considered in any safety analyses because the pipe stub and sensor configuration allows for sufficient time for venting before the actual injection path begins to void. The staff finds this acceptable because the alarms trip with less than half the volume of gas that could affect the injection flow performance, the rate of gas accumulation is expected to be sufficiently slow, and there are no credible postulated gas intrusion mechanisms for these locations.

In its August 20, 2010, response to RAI-DCP-CN66-SRSB-03, the applicant stated that the proposed instrumentation switches provide pre-event operability confirmation and are not used for reactor trip, safeguards actuation, or post-accident monitoring functions. As such, these switches are not required to be part of ITAAC or TS. Also, since the switches are nonsafety related process instrumentation, there is limited periodic calibration or functional checking requirements. The staff considers acceptable the response that the process switches do not meet the requirements to include them in ITAAC and TS.

The proposed changes also included the addition of two new actions and one SR to TS 3.5.6, TS 3.5.7, and TS 3.5.8 related to one or two IRWST injection line(s) "inoperable due to presence of noncondensible gases" and surveillance of the new sensors indication. Also, the appropriate TS Bases would be revised to reflect this change. The staff reviewed the new actions, SR, and bases, which provide for conditions, required actions, completion times, and SRs. The staff found that these parameters conform to guidance provided in the STS and reflect the IRWST injection line gas intrusion and accumulation process analysis described in DCD Section 6.3. Therefore, the staff finds the TS change, as modified in the response to RAI-DCP-CN66-SRSB-03 noted above, to be acceptable.

In addition, the proposed change would provide for system surveillance and venting procedures, as described in DCD Chapter 13, to include inspection of the passive safety system location equipped with manual vent valves to eliminate any identified gas accumulation. Procedural inclusion of vent component inspection is acceptable to the staff because timely inspections could reduce intrusion and accumulation of gases in the injection lines.

In accordance with NUREG-0800 Section 6.3, the staff's evaluation of these proposed design changes must ensure that compliance with GDC 27 and GDC 35 is satisfied. As stated in NUREG-0800, meeting the requirements of:

> ...GDC 27 for the ECCS augments the protection for the primary fission product barrier by providing a means to ensure that the core, under postulated accident

> conditions, can be safely shut down and will be maintained in a coolable geometry.
>
> ...GDC 35 ensures that the ECCS, assuming a single failure, can provide core cooling under accident conditions sufficient to maintain the core in a coolable geometry and to minimize the reaction of water with the fuel cladding.

The staff evaluation of the proposed change concludes that the reliability and performance of PXS is improved. The requirements of GDC 27 and GDC 35 are satisfied based on the following:

- TS requirements that ensure that the PXS is operated in safe condition;
- Adequate procedural inspections to identify and eliminate gas intrusion and accumulation; and
- Sufficient monitoring and alarming features in the control room to ensure timely venting.

23.L.4 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes to provide venting of noncondensible gases in the injection lines enhance the reliability and performance of the PXS to ensure that the core, under postulated accident conditions, can be safely shut down, maintain a coolable geometry, and minimize the reaction of water with the fuel cladding, thus satisfying the requirements of GDC 27 and GDC 35. Therefore, the proposed design changes are acceptable.

23.M Changes to Squib Valve Actuation Time

23.M.1 Description of Proposed Changes

In letters dated April 28, 2010, July 29, 2010, and August 12, 2010, the applicant proposed design changes to the PMS and diverse actuation system (DAS) controls for PXS IRWST injection squib valves actuation by incorporating a 5-second time delay in PXS actuation control for the automatic (and manual) actuation circuitry between the firing of the first and second valve for each pair of squib valves in the same process line. These proposed design changes address the unacceptable design stress on DVI line piping during simultaneous actuation of squib valves in the parallel configuration. The applicant also made similar changes to PMS controls for PXS containment sump recirculation squib valves in parallel path of the same process line. These proposed design changes revise functional diagrams in DCD Figure 7.2-1 by adding Note 5 in Sheet 16, "In-containment Refueling Water Storage Tank Actuation," and Note 6 in Sheet 20, "Diverse Actuation System Logic, Manual Actuation," which state that, for redundant components in a parallel configuration, the components use different time delays to prevent simultaneous actuation.

23.M.2 Regulatory Basis

GDC 35 in 10 CFR Part 50, Appendix A, requires that an ECCS be provided to transfer heat from the reactor core following any LOCA at a rate such that: (1) fuel and clad damage that could interfere with continued effective core cooling is prevented; and (2) clad metal-water

reaction is limited to negligible amounts. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," specifies that the calculated ECCS cooling performance be shown to comply with the acceptance criteria specified in 10 CFR 50.46(b).

The regulatory basis for evaluating the changes in squib valve actuation time is documented in Chapters 6, 7, and 15 of NUREG-1793.

Reviews of the proposed changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes should also conform to the guidelines of Institute of Electrical and Electronic Engineers (IEEE) Standard 7-4.3.2, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by RG 1.152, "Criteria for Use of Computer in Safety Systems of Nuclear Power Generating Stations," and the staff requirements memorandum (SRM) on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.3, "Engineered Safety Features Systems," and Section 7.8, "Diverse Instrumentation and Control Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified in NUREG-0800 Sections 7.3 and 7.8 (as applicable).

23.M.3 Evaluation

The proposed changes add notes in the PMS and DAS functional diagrams in DCD Figure 7.2-1 to state that the redundant components in a parallel configuration use different time delays to prevent simultaneous actuation from adversely impacting accident timing in the safety analyses. For the automatic depressurization system (ADS) stage-4 squib valves, DCD Table 15.6.5-10 specifies the actuation time delay of 60 seconds between the ADS-4A and ADS-4B squib valves. The proposed change was only required for the IRWST injection and containment recirculation sump squib valves since the ADS stage-4 valves already have a 60-second time delay difference in their automatic actuation circuitry.

In its August 12, 2010, response to RAI-DCP-CN08-SRSB-01, the applicant stated that its proposed 5-second time delay for the actuation of redundant squib valves in each IRWST injection path and containment recirculation flow path, respectively, was included not to correct the system safety performance, but rather to protect the integrity of components, piping and structural and mechanical modules, in relationship to the specific squib valves actuation characteristics. The applicant performed an evaluation of LOCA safety analyses with the 5-second delay and found that the impact is not significant. The AP1000 design has the squib valves actuate a relatively long time before they are needed, on the order of several hundreds of seconds, so that the very short squib valve actuation delay of 5 seconds or reasonably longer time for the second squib valve in each path is not significant to the IRWST injection and/or containment recirculation performance of the plant events. Therefore, the safety analyses have not been revised to implement this 5-second time delay. The staff agrees with this explanation. The existing safety analyses in DCD Section 15.6.5 for the large-break LOCA, small-break LOCA, and long-term cooling remain valid and in compliance with GDC 35 and 10 CFR 50.46. Therefore, this proposed change is acceptable.

As documented in NUREG-1793, the staff reviewed and approved Functional Diagram Sheet 16 of Figure 7.2-1 for PXS IRWST injection and containment recirculation isolation squib valves actuation and Functional Diagram Sheet 20 of Figure 7.2-1 for DAS manual actuations for IRWST injection squib valves. The staff reviewed the applicant's proposed design changes to

the two functional diagrams using the review procedures described in NUREG-0800 Sections 7.3 and 7.8.

10 CFR 52.47 requires that an application include a sufficient description and analysis of the SSCs of the facility, with emphasis upon performance requirements, the bases, with technical justification; therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluation. The design information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. Hence, the staff issued RAI-DCP-CN08-ICE-02, requesting the applicant provide clarification for the time delay inconsistency between the design change description and marked-up diagrams and also justify the need for a 5 second time delay between the firing of the two IRWST injection squib valves. In the RAI response, the applicant explained that the 5-second time delay is required to prevent a negative structural impact on the supporting bracket from induced vibrations if both squib valves are fired simultaneously. The firing of one of the commonly mounted explosive valves followed by a five second time delay will have no negative impact on operation. The design change provides an improved protection of the safety function of the IRWST injection and containment recirculation by avoiding potential adverse consequences related to the simultaneous firing of the two squib valves physically housed on the same structural frame module. The staff finds that the design changes and response to the RAI are acceptable. Therefore, RAI-DCP-CN08-ICE-02 is considered resolved. In a subsequent revision to the AP1000 DCD, the applicant made changes to the DCD text which are consistent with the proposed changes.

23.M.4 Conclusion

The staff concludes that the proposed changes of adding notes to the PMS and DAS function diagrams in DCD Figure 7.2-1 are acceptable because they have negligible effect on the LOCA safety analysis, and continue to comply with GDC 35 and 10 CFR 50.46.

In addition, the staff concludes that the applicant's proposed design changes are acceptable because they have no negative impact on operation, and provide improved protection of the integrity of components, piping and structural and mechanical modules.

23.N Changes Related to Anticipatory Reactor Trip in the Event of an Inadvertent Passive Residual Heat Removal Actuation

23.N.1 Description of Proposed Changes

In a letter dated May 10, 2010, and in two letters dated July 29, 2010, the applicant proposed design changes related to anticipatory reactor trip in the event of an inadvertent PRHR actuation. The applicant proposed design changes to the PMS to include one additional reactor trip to mitigate an inadvertent PRHR event caused by the opening of either of the two PRHR HX discharge valves when the reactor is in power operation. In the PMS logic control system, a reactor trip that is to be added will be generated when either of the two PRHR HX discharge valves comes off its fully shut seat while the reactor is at power. Another proposed design change is to adjust the frequency of the inservice test (IST) for the two PRHR HX discharge valves from the current once every quarter to every cold shutdown.

In order to mitigate an inadvertent PRHR actuation event, the applicant proposed changes to add a PRHR actuation reactor trip function, which is based on the PRHR HX control valve (CV) indication, to DCD Chapter 7.2, "Reactor Trip." Related to this design change, a sentence is added to DCD Section 6.3.7.6.1, which states that for the PRHR HX discharge valves, valve position indication is used to initiate a reactor trip upon opening of these valves while the reactor is at power. Also, DCD Section 15.1.6, "Inadvertent Operation of the PRHR Heat Exchanger," is revised to state that to prevent the reactivity increase (as a result of inadvertent actuation of the PRHR HX event) from causing reactor power increase, a reactor trip is initiated when either PRHR discharge valve comes off of its fully shut seat. DCD Section 15.1.6.2 is revised to state that since a reactor trip is initiated as soon as the PRHR discharge valves are not fully closed, this event is essentially a reactor trip from the initial condition and requires no separate transient analysis. In DCD Table 15.0-4a, the reactor trip function on the PRHR discharge valve not closed with a time delay of 1.25 seconds is added, and in DCD Table 15.0-6, this reactor trip function is listed for the inadvertent operation of the PRHR event.

23.N.2 Regulatory Basis

The regulatory basis for evaluating the proposed changes to the AP1000 PMS reactor trip system is documented in Chapters 7, 15, and 16 of NUREG-1793.

Reviews of the changes are also based on meeting the relevant requirements of regulations and guidelines in 10 CFR 50.55a(h); 10 CFR 52.47; and GDC 20 and GDC 21 of 10 CFR Part 50, Appendix A. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.2, "Reactor Trip Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified in NUREG-0800 Section 7.2 (as applicable).

GDC 10 in 10 CFR Part 50, Appendix A, specifies that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. An inadvertent operation of the PRHR heat exchanger is an AOO and, therefore, must comply with GDC 10.

23.N.3 Evaluation

As documented in NUREG-1793, the staff reviewed and approved the reactor trip system as specified in Section 7.2 of AP1000 DCD, Revision 15. The staff reviewed the applicant's design changes to the reactor trip system using the review procedures described in NUREG-0800 Section 7.2.

If either of the two PRHR HX discharge valves inadvertently comes off its normally fully closed seat while the reactor is at power, it allows a slug of cold water to be introduced into the reactor core through the PRHR HX to cause a marked increase in reactor power, which may exceed the fuel design limits. Therefore, a reactor trip is needed to mitigate this inadvertent PRHR event introduced by the opening of either of the two PRHR HX discharge valves, and prevent the fuel design limits from being exceeded. However, in the event of such an inadvertent PRHR event, the current reactor trip design included in the AP1000 DCD, Revision 17 will not be able to function fast enough to mitigate this inadvertent PRHR event. Therefore, the proposed design changes result in an improved reactor protection system to mitigate the inadvertent PRHR event. In addition, the 2-out-of-4 control logic with bypass capability is also used for this reactor trip condition in this DCP. The staff finds that all those changes meet applicable requirements in

10 CFR 50.55a(h), GDC 20, and GDC 21. The applicant made necessary changes to DCD Tier 1 Section 2.5.2, Tier 2 Section 7.2, and other related tables and figures. The staff finds that the proposed changes to reactor trip logic for an inadvertent PRHR actuation are acceptable because they satisfy the requirements of 10 CFR 50.55a(h), GDC 20, and GDC 21. In a subsequent revision to the AP1000 DCD, the applicant made changes to the DCD text, which are consistent with the proposed changes.

An inadvertent operation of PRHR HX causes an injection of relatively cold water into the RCS, resulting in a reactivity insertion due to a negative moderator temperature coefficient. Currently, several reactor trip functions are available to mitigate the event, including the overpower ΔT and overtemperature ΔT trip functions, to prevent a power increase, which could lead to a DNBR less than the safety analysis DNBR limit. The safety analysis of the limiting inadvertent operation of PRHR HX, presented in DCD Section 15.1.6, shows that, without taking credit for a reactor trip function, the nuclear power rises to about 120 percent temporarily, then drops and reaches a new equilibrium condition at about 108 percent of the nominal value. The RCS pressure and minimum DNBR are within the respective limits.

The proposed design change to add a PRHR actuation reactor trip function will enhance the reactor protection to trip the reactor upon event initiation. In its July 29, 2010, response to RAI-DCP-CN60-SRSB-01, the applicant stated that it has performed an evaluation of the operation of the PRHR HX transient assuming a conservative 1.25 second reactor trip response time, which covers the time to sense the opening of a PRHR valve through initial insertion of control rods. A confirmatory analysis using LOFTRAN shows that the minimum time between the opening of the PRHR valves and any colder water reaching the reactor core inlet is at least 2 seconds. Since the control rods are already inserting into the core before any colder water reaches the core inlet, there will not be an adverse power increase for this transient. Since an inadvertent operation of the PRHR HX event can only be caused by an inadvertent opening of the PRHR HX discharge valves (either by operator error, false actuation signal, or malfunction of a discharge valve), and a reactor trip will be initiated as soon as a PRHR discharge valve comes off the closed seat position, the consequence is mitigated as soon as the inadvertent PRHR HX operation event occurs. The proposed design changes continue to comply with SAFDL and no additional safety analysis is needed.

Each of the two PRHR HX discharge valves has one set of existing Class 1E magnetic valve position indicators (VPIs), one for the open position indication and one for the close position indication. The design change will use the existing close position indicator and add three more Class 1E magnetic VPIs to give a total of four closed signals per valve. This configuration is necessary for the "two out of four" logic required for a reactor trip signal. The proposed change in DCD Section 6.3.7.6.1 merely indicates that the PRHR HX discharge valves' position indication is used for the reactor trip function, and is, therefore, acceptable.

The applicant's proposed design changes add a reactor trip signal generated from the opening of valves in the PRHR system. This trip would minimize the effect of the anticipated reactivity excursion due to cold water addition by the PRHR system initiating flow. This change affected the TS for reactor trip system instrumentation, Table 3.3.1-1, as well as discussions in bases Sections B 3.1.1 and B 3.3.1. The additional information included in this change added appropriate descriptions of the trip function and purpose, and created a trip function in Table 3.3.1-1 that provides adequate protection against this event. The staff finds this change acceptable since it conforms to the format and trip functions described in the STS.

The applicant's proposed design changes also include changing the inservice testing full-stroke exercising requirement for the PRHR HX valves PXS-V108a and PXS-V108b from once a quarter to every cold shutdown. Valves PXS-V108a and PXS-V108b are Class 1, Category B air-operated valves that function as discharge valves for the PRHR HX. The inadvertent actuation of the PRHR HX causes an injection of relatively cold water into the RCS. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. To prevent this reactivity increase from causing reactor power increase, a reactor trip is initiated when either PRHR discharge valve comes off its full shut seat. The staff considers changing the IST full-stroke exercising requirement for valves PXS-V108a and PXS-V108b from once a quarter to every cold shutdown to be acceptable based on the ASME Code (2004E) ISTC-3521(c), which states that if exercising is not practicable during operation at power, it may be limited to full-stroke testing during cold shutdown.

23.N.4 Conclusion

After reviewing the proposed changes and markups to associated DCD Tier 1 and Tier 2 sections, tables, figures, and TS, the staff finds that the proposed changes provide an improvement to the reactor protection system. The staff concludes that the proposed changes to add a PRHR actuation reactor trip function is acceptable because it will trip the reactor as soon as an inadvertent operation of PRHR HX event occurs, thereby ensuring that the SAFDL will not be exceeded. Therefore, the staff concludes that these proposed design changes are acceptable.

23.O Changes to Reactor and Turbine Trips Functional Logic of Diverse Actuation System

23.O.1 Description of Proposed Changes

In letters dated May 20, 2010, and July 29, 2010, the applicant proposed design changes to add a reactor trip and turbine trip to the functional logic of the DAS based on the 2-out-of-2 control logic of the PRHR high hot leg temperature sensor outputs.

23.O.2 Regulatory Basis

The regulatory basis for evaluating these proposed design changes of DAS PRHR high hot leg temperature logic is documented in Chapter 7 of NUREG-1793.

Reviews of the changes are also based on meeting the relevant requirements of regulations and guidelines in 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of GDC 13 and GDC 20 in 10 CFR Part 50, Appendix A. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.8, "Diverse Instrumentation and Control Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified in NUREG-0800 Section 7.8 (as applicable).

23.O.3 Evaluation

As documented in NUREG-1793, the staff reviewed and approved the DAS in AP1000 DCD Tier 1 Section 2.5.1 and Tier 2 Section 7.7.1.11. The DAS system in the certified AP1000 DCD, Revision 15, uses the 2-out-of-2 control logic, which is based on the PRA evaluation. The staff

reviewed the proposed design changes using the review procedures described in NUREG-0800 Section 7.8 and applicable regulations of GDC 13 and GDC 20 in 10 CFR Part 50, Appendix A.

According to the modeling in the PRA, the original DAS design in the AP1000 DCD should initiate a reactor trip and turbine trip for anticipated transients without scram (ATWS) sequences with the main feedwater available. Since the main feedwater is still available, the low SG water level signal will not be generated to initiate the reactor or turbine trip in the DAS. Therefore, the high hot leg temperature signal is needed in the DAS to trip the turbine or the reactor via the control rod motor-generator (MG) sets.

10 CFR 52.47 requires that an application include a sufficient description and analysis of the SSCs of the facility, with emphasis upon performance requirements, the bases, with technical justification; therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluation. The design information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. Hence, the staff issued RAI-DCP-CN63-ICE-01 requesting the applicant provide justification for adding another time delay for opening PRHR discharge valves, which is not explained in the DCP. The applicant's July 29, 2010, RAI response stated that the time delay has been provided in the functional design to support sequencing of the output field devices associated with a system-level control function. This delay time is added to be consistent with the use of timers in the existing functional design (e.g., low SG water level logic). The staff finds that the response to the RAI is acceptable.

After reviewing the proposed design changes and RAI responses, the staff finds that the proposed design changes are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.O.4 Conclusion

After reviewing the proposed design changes, including markups to DCD Tier 1 and Tier 2 and RAI responses, the staff finds that the proposed changes will meet the required DAS functions as modeled in the PRA and related regulatory criteria. Therefore, the staff concludes that these proposed design changes are acceptable.

23.P Changes to Steam Generator System Instrument Piping

23.P.1 Description of Proposed Changes

In a letter dated July 8, 2010, the applicant proposed design changes in the material for the SGS instrument piping. These design changes would modify Figure 10.3.2-1 of the AP1000 DCD to specify stainless steel piping for all AP1000 Quality Classes B and C instrument piping for the SGS. AP1000 Quality Classes B and C are designed and fabricated to ASME Code, Section III, Class 2 and 3, respectively. The main steam supply system (MSSS) includes the AP1000 SGS.

23.P.2 Regulatory Basis

The staff reviewed and evaluated the proposed design changes in accordance with the guidance in NUREG-0800 Section 10.3.6 to ensure that the ASME Code, Class 2 and 3 MSSS

components use material specified in Sections II and III of the ASME Code, thereby meeting the requirements of GDC 1, "Quality Standards and Records," and 10 CFR 50.55a.

23.P.3 Evaluation

Figure 10.3.2-1 modified all of the ASME Code, Section III, Class 2 and 3 SGS instrument lines, which typically are 2.5 cm (1 in) NPS and less, to be a corrosion resistant (stainless steel) material. Section 10.3.6.2 of the AP1000 DCD specifies that the material selection and fabrication for ASME Code, Section III, Class 2 and 3 are in accordance with the requirements of ASME Code, Section III, Class 2 and 3 components outlined in Sections 6.1.1.1 and 6.1.1.2 of the AP1000 DCD. Section 6.1.1.1 of the AP1000 DCD specifies that pressure-retaining materials meet the requirements of Articles NC-2000 and ND-2000 of the ASME Code, Section III for Class 2 and 3 components, respectively. The staff notes that Articles NC-2000 and ND-2000 for ASME Code Classes 2 and 3, respectively, specify that the material be in accordance with Section II of the ASME Code. Based on this information, the staff finds that the use of stainless steel material specified in ASME Code, Section II for ASME Code Classes 2 and 3 SG instrumentation piping, typically 1 NPS or less, is acceptable. The staff's acceptance is based on the use of stainless steel for the instrumentation piping, which is more corrosion resistant than carbon steel, and that the material will be procured in accordance with ASME Code Section II and fabricated in accordance with ASME Code Section III. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.P.4 Conclusion

The staff concludes that the proposed use of stainless steel instrument piping for the SGS provides higher corrosion resistance than carbon steel, and that the material will be procured and fabricated in accordance with ASME Code Sections II and III as specified in the guidance of NUREG-0800 Section 10.3.6 and, therefore, meets the requirements of GDC 1 and 10 CFR 50.55a. The staff concludes that the proposed design changes are acceptable.

23.Q Changes to the Steel Containment Vessel Girder and Polar Crane Rail Clip

23.Q.1 Description of Proposed Changes

In letters dated April 26, 2010; August 26, 2010; and September 16, 2010, the applicant proposed design changes to the steel containment vessel (SCV) girder and polar crane rail clip. These design changes apply to the safety-related structure (containment vessel) and the heavy load handling system (heavy load lifting equipment and support) and include the following changes:

- Increase in the thickness of the girder top plate from 3.8 cm (1.5 in) to 4.45 cm (1.75 in)
- Change in the rail support from Gantrex Pad to a bolted clip design
- Extension of the SCV girder inward by 6.98 cm (2-3/4 in)

23.Q.2 Regulatory Basis

Sections 3.8.2 and 9.1.5 of NUREG-1793, address the function and acceptability of the SCV associated support structures for the overhead heavy load handling system in the AP1000 standard design. Acceptability of the proposed design change is evaluated to determine

whether the design meets the relevant requirements specified in 10 CFR Part 50, Appendix A, GDC 2, which requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

23.Q.3 Evaluation

The SCV associated support structures are designed to provide the necessary support for the overhead heavy load handling (polar crane) system in the AP1000 standard design. The polar crane systems are safety-related and are used to handle heavy equipment, such as the integrated head package, SGs, and HXs. As stated in the DCD changes included in this design change, the design of the SCV associated support structures for the polar crane system support is consistent with the SSE design of equipment anchorages of seismic Category I equipment. The girder top plate and the new bolted clip design for the rail support are analyzed to show that the structure can withstand an SSE event.

During the staff's evaluation of the proposed design changes to the SCV associated support structures, it was determined that additional information was required from the applicant to complete the staff's evaluation. The staff generated RAI-DCP-CN-5-SEB-01 requesting: (1) an analysis including materials and design basis loading used to demonstrate that the increase of 0.64 cm (1/4 in) from 3.81 cm (1-1/2 in) to 4.45 cm (1-3/4 in) thick in top plate of the crane girder is adequate to support the design-basis loadings including a seismic load; (2) an analysis to demonstrate the new rail clip design, including clip spacing is sufficient to meet the design-basis load requirements including the seismic load demands; and (3) the basis for determining the extent of the inward extension of the CV girder top plate.

The applicant's response to RAI-DCP-CN-5-SEB-01 provided sufficient information on the design analysis for the proposed design changes to support the staff's evaluation. In addition to evaluating the applicant's proposed design changes and RAI responses, the staff performed two audits.

With respect to the SCV polar crane girder top plate design, the applicant presented a design analysis, which is documented in the technical report APP-MV50-S2C-020, Khanh Do, "Polar Crane Girder Top Plate Analysis." The stress analysis considers a total of 192 load combinations. Using ANSYS 11.0, a commercially available general purpose code, a 3D finite element model was constructed with the boundary conditions fixed at Elevation (El.) 30.5 m (100 ft) (i.e., at the grade level) for the purpose of determining the maximum stress intensities at the top plate. It was found that the maximum stress intensity [

] is well within the allowable stress intensity of 90 kips per square inch (ksi) (620.5 MegaPascal (MPa)) according to ASME NE-3221 in ASME Code Section III, Division 1, Subsection NE, Class MC, 2001 Edition. Accordingly, a 4.45 cm (1.75 in) thick plate is shown to be adequate to support the design-basis loadings, because the maximum stress intensity generated in the top plate is less than the allowable stress intensity the steel plate can offer. The staff reviewed the methodology of the analysis including model construction, boundary and loading conditions, material properties and applicable code; and determined that the design analysis is acceptable, and the use of a 4.45 cm (1.75 in) thick top plate is in compliance with the ASME Code requirements.

With respect to the new rail clip design with the bolted clips, the applicant presented an analysis in PR-08-5020/70587483, Roger Johnson, "Polar Crane Mechanical Calculations," Revision 1,

March 5, 2010, to show that the proposed new design is capable of resisting a horizontal wheel load [ ] due to seismic conditions. With the help of MathCad Version 13 and GTSTRUDL 29.1 seismic analysis, it was found that the rated load [ ] governs the main hoist system, whose design meets NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" requirements. The new clip design is given in Westinghouse drawing APP-MH01-V2-021 and Westinghouse drawing APP-MH01-V6-041 and is based on the adopted 76.20-cm (30-in) wheel diameter. [

] With the use of the clip plate [ ] with three slotted holes, the maximum bending stress [ ] is less than the 81 ksi allowable specified by the code. The clip plate is bolted [ ] to the filler plate welded to the girder top plate [

] and the induced shear [ ] will be within the allowable shear stress of 148.9 MPa (21.6 ksi) based on the American Welding Society (AWS) code (the ASME NOG-1 code allowable is 50 percent higher, i.e., 223.4MPa (32.4 ksi)). The staff reviewed the detailed calculations and found the new design as given in drawings APP-MH01-V2-021 and APP-MH01-V6-041 to be in compliance with the AWS and ASME Codes; thus, it is acceptable to the staff.

The third item of design changes is concerned with the inward extension of 6.98 cm (2-3/4 in) of the CV girder top plate. The reasoning, according to the submittal, is that the tolerance requirement of the polar crane rail is much tighter than the SCV girder. The staff requested the applicant provide the basis to justify that the increase of 6.98 cm (2.75 in) inward of the SCV girder top plate is quantitatively adequate to meet the tolerance requirement. In its response, the applicant provided ASME Code, ASME-NOG-01-1998 (NOG-1) as the basis in the polar crane design documented in Westinghouse calculation APP-GW-GEE-513, DCP3513, "Steel Containment Vessel Girder and Polar Crane Rail Clip Design." The code NOG-1 specifies that the maximum and minimum span between the rails must be no greater or less than the nominal span of the rails +/- 0.76 cm (3/8 in). This tolerance is tighter than the girder, based on construction experts, who specify the tolerance for the radius of the SCV girder to be +/- 4.95 cm (1.95 in). Because of this difference in tolerance requirements, the girder, including top plate, web and bottom plate, must be extended radially inward toward the center of the CV. Figure 5 in Westinghouse calculation APP-GW-GEE-513 provides the new design showing the tolerance stack-up and the required extension of the top plate. Figure 1, Note B and Figure 2 show the associated design changes. The staff reviewed the drawings and the new design, and found the design changes are in compliance with the ASME Code NOG-1, and thus, are acceptable.

The proposed design changes, including DCD markups and RAI responses, are acceptable. RAI-DCP-CN-5-SEB-01 is resolved.

23.Q.4 Conclusion

The staff concludes that the proposed design changes are acceptable because the proposed changes will not adversely affect safety-related SSCs and the capability of the polar crane systems to perform their intended functions of heavy load lifting and transportation. The proposed design changes were evaluated with respect to conformance with the existing AP1000 licensing basis and found acceptable.

23.R Changes to the Reactor Vessel Support System

23.R.1 Description of Proposed Changes

In letters dated May 10, 2010, and September 9, 2010, the applicant proposed design changes related to the structural support system for the RV, which is a safety-related structure. These design changes involve the following modifications of the RV support system:

- Eliminate reliance on the CA04 structural module as part of the reactor pressure vessel (RPV) support system.
- Provide support boxes as RPV supports
- Change the anchor supports from a "bolted into embedded plates" configuration to a "anchored directly to primary shield wall concrete base via steel embedment plates" configuration.
- Increase the length of the RPV support boxes or legs.
- Install wear plates for the RPV bearing and tribological performance (i.e., to support RV and reduce frictional wear due to thermal expansion).

23.R.2 Regulatory Basis

Sections 3.8.3 and 5.4.10 of NUREG-1793, address the function and acceptability of the RPV support structural box in the AP1000 standard design. Acceptability of the design change was evaluated for its conformance with the existing AP1000 licensing basis.

23.R.3 Evaluation

In the AP1000 standard design, the RPV structural support system is designed to provide the necessary support for the RPV. The original anchorage design was bolting into embedded plates of the CA04 structural module. Design finalization analyses determined that the resulting stresses exceed the allowable stresses. As a result, the applicant proposed design changes in which the overstressed CA04 module is not used to support the RPV. Instead, the applicant has proposed a revised design where four support boxes are used to support the RPV. There are four support "boxes" or "legs" located at the bottom of the RPV's cold leg nozzles. The support boxes are anchored directly to the primary shield wall concrete base via steel embedment plates. As stated in the proposed design changes, the design of the RPV associated support structures is consistent with the SSE design of seismic Category I equipment. The RPV support boxes, including the new anchorage design, are analyzed to show that they can withstand an SSE event.

During the staff's evaluation of the proposed design changes to the RPV associated support structures, it was determined that additional information was required from the applicant. The staff requested additional information related to: (1) a stress analysis of the proposed design, including materials and design basis loading used to demonstrate that the maximum stresses incurred at the critical sections of the RPV support box structure are lower than those of the previous design, and that the new design is in compliance with the adopted code requirements; (2) specifications of the wear plates, including material, lubricant used, geometric dimensions in

size and shape, and performance requirements; and (3) an assurance of satisfactory tribological performance that uneven settlement on the plate's top surface will not deter lateral movement of the RV due to thermal expansion, and that the wear plate can endure the frictional wearing due to cyclic motion of the heavy RPV over its entire design life without loss of its intended design function.

In a letter dated September 9, 2010, the applicant provided sufficient information on the design analysis to support the staff's evaluation. The supporting information also included documentation on the material specifications of the self-lubricating bearings used for the wear plates.

With respect to the RPV support box design, the applicant stated that the design is in compliance with the design specification depicted in APP-SS30-Z0-001, "Design Specification," Revision 1. Thus, the support box design meets the requirements of ASME Code, Section III, Subsection NF, 1998 Edition, up to and including the 2000 Addenda. The support structure is classified as NF Class 1. Allowable stress limits of ASME Code NF-3220 for the Level A, B, C and D load combinations were calculated for the material used [ ] and it was found that the Level D loading condition governs the design. For design analysis, a three-dimensional finite element analysis (FEA) model was constructed using the general purpose FEA computer code ANSYS. Details of the methodology including the modeling techniques, boundary conditions and loading input are documented in the report APP-PH01-Z0C-007, "Finite Element Analysis of the RV Support Structure," Revision 0, May 26, 2010. The resulting stresses, including linearized primary membrane (P_m) and primary membrane plus bending ($P_m + P_b$) stress intensities, were computed from the FEA model. [
] Thosee resultant stress intensities were compared with the allowable stress intensities. The comparisons demonstrate that, for the Level D loading condition, [
]. The staff performed audits of a supporting document on design specifications (APP-SS30-Z0-001) and a supporting document on FEA (APP-PH01-Z0C-007). These audits confirmed that the information in the supporting documents was consistent with the applicant's proposed design changes. Therefore, the staff finds the proposed design changes to be acceptable because they satisfy the ASME Code requirements.

In its September 9, 2010, letter, the applicant stated that the RPV support bottom and side wear plates are specified as Lubron wear plates fabricated by Lubron Bearing Systems. [

] Detailed specifications of the wear plates used were provided by the applicant. Specifications of the bottom and side wear plates are provided in AP1000 RPV support design drawings APP-PH01-V2-211 (General Assembly) and APP-PH01-V2-212 (Component Details). During RPV installation, the bottom wear plate and interfacing thermal plate are assembled with a bluing process to insure a high degree of uniform contact (>75 percent) between mating surfaces. By design, these wear plates are good for up to 55.2 MPa (8,000 psi) bearing pressure and up to 593 °Celsius (C) (1,100 °Fahrenheit (F)). The actual bearing pressure for the AP1000 RPV support for the Level A service condition is approximately 15.9 MPa (2,300 psi), much less than the bearing capacity of 55.2 MPa (8,000 psi). The RPV support, its connection to the foundation, the foundation, and the wear plate connection to the support have all been designed for friction loads during normal plant heat-up and cool down thermal cycles. The staff's review also considered the potential for uneven settlement of the RPV during

installation. Tier 1 Table 2.1.3-2 specifies the ITAAC for the reactor system, which includes the RPV. Several ITAAC exist to address this issue, and include acceptance criteria that a report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions. The staff finds that the proposed design changes are acceptable in terms of structural and tribological performance.

23.R.4 Conclusion

The staff concludes that these proposed design changes are acceptable because the proposed changes will adversely affect neither safety-related SSCs, nor the capability of the RPV support boxes to perform their intended function of supporting the RPV while allowing free lateral movement of the four legs due to thermal expansion. These design changes were evaluated with respect to conformance with the existing AP1000 licensing basis and found acceptable.

23.S Changes to the Passive Containment Cooling System

23.S.1 Description of Proposed Changes

The applicant has proposed changes to the shield building from the Revision 15 DCD to address additional external hazards and to simplify construction. The applicant assessed the impact of these changes to the PCS in APP-GW-GLR-096, "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analysis," Revision 1, and submitted the report to the NRC in a letter dated August 6, 2010.

This report includes a description of the enhanced shield building design changes, a discussion of how these changes impact design basis test results, and evaluations of the limiting DBA and beyond design basis accidents (BDBA). Appendix A of APP-GW-GLR-096 includes proposed DCD changes. The changes that impact NUREG-1793 Sections 6.2.1, "Containment Functional Design"; 6.2.2, "Containment Heat Removal Systems"; and 6.2.3 "Shield Building Functional Design"; are as follows:

- Update containment response to reflect results of an analysis that incorporates the enhanced shield building design.
- Lower the required reactor decay heat limit for air only cooling.
- Specify the long-term makeup rates the PCS must simultaneously provide to containment and the SFP when the plant is refueling.

These proposed design changes also include revisions to several TS. In TS Table 3.3.2-1, "Engineered Safeguards Actuation System Instrumentation"; TS Table 3.3.5-1, "DAS Manual Controls"; and TS 3.6.7, "Passive Containment Cooling System (PCS) – Shutdown"; the Modes 5 and 6 applicability is revised to reflect the lower reactor decay heat limit for air only cooling mentioned above. In TS 3.7.9, "Fuel Storage Pool Makeup Water Sources," the notes associated with LCO 3.7.9, which list special plant conditions for each available makeup water source, are revised to account for changes to TS 3.6.7 regarding the availability of the passive containment cooling water storage tank (PCCWST) as a makeup water source. The cask loading pit (CLP) is added as a third makeup water source in addition to the PCCWST and the cask washdown pit. Two new surveillance requirements are also added to ensure readiness of water inventory from the PCCWST and the CLP, when needed.

In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. Proposed changes to Chapter 9 are evaluated in Section 9.1.3 of this report.

23.S.2 Regulatory Basis

The following Commission regulations are related to the evaluation of the enhanced shield building:

- GDC 16, "Containment Design," as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require
- GDC 38, "Containment Heat Removal," as it relates to the ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a LOCA and to maintain them at acceptably low levels
- GDC 50, "Containment Design Basis," as it relates to demonstrating sufficient margin in accident analysis
- 10 CFR 52.47(c)(2), as it relates to design certification testing in support of a passive plant design

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the PCS and the SFP design and operating information described in DCD Sections 6.2 and 9.1.3 respectively. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance in NUREG-0800 Chapter 16.

23.S.3 Evaluation

One of the safety-related functions of the PCS is to provide air flow over the outside of the containment vessel by means of a natural circulation air flow path. Air enters through inlets located at the top of the shield building and then travels downward into the annulus between the inside of the shield building and the air baffle. At the bottom of the baffle wall, the air turns 180 degrees into the riser annulus formed by the baffle and the outside of the containment vessel. Air continues past the top of the containment vessel and exits through the shield building chimney.

In the Revision 15 design, air is admitted into the top of the shield building through 15 large, uniformly distributed openings, with fixed louvers and screens to prevent weather and wildlife from entering the shield building. In the modified design, a steel structure containing 29 uniformly distributed openings with fixed louvers and screens is added around the top outside of the shield building. The flow area in the steel structure is slightly higher than the flow area through the louvered and screened inlet openings in the previous shield building design. Air enters through these openings, collects in the plenum created by the structure and outside of the shield building, and then enters the shield building through 236 inlet ducts. These shield building inlets have a much lower flow area than the louvered and screened inlet openings in

the previous shield building design. The enhanced shield building chimney is shorter than the original design; therefore, the buoyant driving head for the PCS flow is expected to decrease. In the enhanced design, two heavy steel grates are added within the chimney region to protect the containment shell from external hazards, which will increase the discharge flow resistance and reduce the discharge flow area.

The NRC approved the functional design of the containment systems for the AP1000 as described in NUREG-1793. The agency based its approval on data from a number of test facilities that were specifically designed to model processes and phenomena of the AP1000 containment design following a major piping rupture within the containment building. These data were used for the qualification of the WGOTHIC computer code, which was then used for the safety analysis of the AP1000. The test series included heat and mass transfer tests, wind tunnel tests, water distribution tests, integral system tests of the PCS, and air flow path characterization tests. All of these tests were to be within the range of the physical conditions predicted for the enhanced shield building design except for the air flow path characterization test which modeled the previous shield building. The flow path pressure drop is expected to increase in the new design; therefore, the applicant repeated the air flow path characterization test on apparatus specific to the revised shield building. The results were used to confirm the local annulus loss coefficients used in the revised WGOTHIC model.

From the phenomena identification and ranking table that was developed for the AP1000 in Westinghouse Commercial Atomic Power (WCAP)-15613, "AP1000 PIRT and Scaling Assessment," issued February 2001, The applicant identified the high-ranked (most important) phenomena and processes for evaluation of the containment response following a LOCA or main steamline break (MSLB). These include condensation on the inside surface of the containment shell, conduction through the shell, and evaporation from the containment shell liquid film that flows down from the PCS water storage tank to the top outside of the containment building. Air flow through the containment shield building annulus is not listed with the high-ranked phenomena.

The staff agrees that air flow through the shield building need not be highly ranked following a LOCA or MSLB, because in these instances, the PCS will release water over the containment shell to provide evaporative cooling, and evaporative cooling is a much more significant heat removal mechanism than air flow convection. Analyses by the applicant and the staff illustrate this conclusion. APP-GW-GLR-096 describes the Westinghouse analysis based on the WGOTHIC evaluation model that was approved by the staff for the Revision 15 DCD analysis and subsequently modified to incorporate the proposed wet bulb temperature increase to 30 °C (86.1 °F) discussed in Section 6.2.1.1.1 of the DCD along with the pressurizer room changes discussed in Section 6.2.1.2 of the DCD.

To address the enhanced shield building design, the applicant further revised the model to include the addition of the PCS air inlet structure, reduction of the flow areas in the shield building inlet and exit, reduction of the shield building chimney height, and increase to the air flow path resistance. The applicant ran the design basis LOCA and MSLB events with the revised model and the results demonstrated that even though the natural circulation air flow decreased, the effect on containment pressure and temperature was insignificant. Additionally, the pressure at 24 hours following a large cold leg break LOCA was unchanged. The staff reviewed the detailed modeling changes during audits on April 21, 2010 and September 3, 2010. The subject of the audits was APP-SSAR-GSC-746, "Containment Response Analysis for the AP1000 Shield Building Design Change." The staff found the estimates on loss coefficients, which were made prior to completion of the air flow path

characterization test, to be reasonable because they bounded predictions [ ] for a thick orifice. Furthermore, when pressure drops from the air flow path characterization test became available, these estimated loss coefficients were demonstrated to be conservatively high.

The staff performed a confirmatory analysis using the CONTAIN computer code with an AP1000 model developed by the staff during the DCD review. The shield building changes were incorporated and used to evaluate a large cold-leg break LOCA, which is the peak pressure DBA. The results were consistent with the Westinghouse evaluation; the PCS air flow decrease had a negligible impact on peak pressure, peak temperature, and containment pressure after 24 hours.

The applicant proposed changes to incorporate the results of its analysis into the DCD tables summarizing postulated accident values, DCD Tier 2, Tables 6.2.1.1-1 and 6.2.1.1-3. The changes to the LOCA and MSLB results were found acceptable because they are consistent with the enhanced shield building design. Changes made to the external pressure results are evaluated in Section 23.W of this report.

The natural circulation air flow is reduced in the enhanced shield building design; therefore, the amount of heat that can be removed during air only PCS operation will also be reduced. As a result, the applicant proposed lowering the reactor decay heat limit for air only PCS operation from 9 megawatts thermal (MWt) to 6 MWt. APP-GE-GLR-096 describes the supporting WGOTHIC analysis demonstrating the containment pressure could be maintained below the design value of 407 kPa (59 psig) for seven days with no PCS water release (and subsequent evaporative cooling) assuming an initial decay heat rate of 6 MWt.

The staff reviewed the analysis basis during its July 27, 2010, audit of APP-SSAR-GSC-749, "AP1000 Dry PCS Heat Removal Capability." The evaluation model was the previously described LOCA model modified to turn off the PCS water, replace the accident mass and energy forcing functions with decay heat input to the IRWST, and adjust the air flow path loss coefficients to reflect a 30 percent increase over the air flow path characterization test results. The staff finds the loss coefficient values reasonable because they are based on physical measurements with added margin to bound uncertainties in the test results.

The staff ran a confirmatory CONTAIN analysis using the previously described LOCA model modified for no PCS water flow and an initial decay heat rate of 6 MWt.

The results were consistent with the applicant's evaluation; the air flow across the containment shell decreased and the containment pressure increased compared to the original shield building, but containment pressure remained below the design value for the run duration of seven days.

In an additional study, the applicant re-evaluated the containment pressure resulting from the BDBA of a prolonged loss of offsite power concurrent with a complete loss of the passive containment cooling water. For this unlikely event, the PRHR HX transfers reactor decay heat and the system sensible heat into the IRWST pool. When the water in the IRWST is heated to boiling, the steam that is released condenses on the containment internal structures. As the internal structures are heated, the containment pressure rises and heat is transferred through the containment shell to the air traveling up through the shield building annulus. The applicant performed the analysis with a "best estimate" WGOTHIC model. The results showed that even with the reduced natural circulation air flow associated with the enhanced containment building,

it will take more than 24 hours for the containment pressure to reach the maximum pressure capability limit of 889 kPa (129 psig) defined in DCD Section 3.8.2. Therefore, the enhanced shield building meets the DCD Section 19.34.2.6 statement that, with air-only cooling, containment failure is predicted to occur more than 24 hours after accident initiation. The staff reviewed the analysis basis during its July 27, 2010, audit of APP-SSAR-GSC-749. The assumptions for the "best estimate" BDBA evaluation model, which included changes to initial temperatures, heat transfer coefficients, and loss coefficients, were found to be reasonable and consistent with evaluations of beyond design basis events.

The PCS has a design commitment to provide containment cooling and SFP makeup simultaneously from post-72 hours to seven days after DBA initiation at the minimum flow rates specified in DCD Tier 2, Table 6.2.2-1. The applicant recognized that the proposed reduction to the maximum reactor decay heat limit for air only containment cooling created a scenario whereby the currently specified minimum SFP supply of 132.5 Lpm (35 gpm) would not be adequate to maintain coverage of the fuel in the SFP. During refueling, when the full core is split such that the reactor decay heat is greater than 6 MWt and the SFP decay heat is less than 7.2 MWt, the PCS water is reserved for containment cooling for the first 72 hours and cannot be used for SFP makeup until after this time. The applicant determined a DBA at this plant condition would require a minimum of 189.3 Lpm (50 gpm) SFP supply. Because the total PCS flow is limited, an increase in the SFP supply requires a reduction to the currently specified containment make-up rate of 378.5 Lpm (100 gpm). The 378.5 Lpm (100 gpm) is based on containment cooling requirements post-72 hours after a limiting DBA occurring at full power. The necessary flow following a DBA during refueling will always be less than this because the reactor must be shut down for 100 hours prior to the start of refueling to provide time for the RCS to cool down and depressurize. The applicant reduced the required flow rate for containment cooling during refueling to 302.8 Lpm (80 gpm). The supporting analysis, APP-SSAR-GSC-750, "WGOTHIC Validation of Post-72 Hour Containment Cooling Flow Rates for Accident Scenarios after Refueling" was audited by the staff on July 30, 2010.

The evaluation considered a loss of power event coincident with the start of refueling, modeled by adding decay heat representative of 100 hours after shutdown to the IRWST to represent the PRHR HX system. The model assumed full PCS flow for the first 72 hours and 302.8 Lpm (80 gpm) flow thereafter. The results demonstrated that the containment pressure remained well below the design limit of 407 kPa (59 psig) for seven days. The staff found the modeling assumptions to be consistent with design basis analysis and the significant margin in the results provided further assurance that this is an acceptable change.

With respect to proposed changes to TS Sections 3.3.2, 3.3.5, 3.6.7 and 3.7.9 and their associated Bases, the staff finds these changes acceptable because they reflect the PCS and the SFP cooling system design and operating information described in DCD Sections 6.2 and 9.1.3, respectively. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.S.4 Conclusion

The staff's review concludes that the design changes described above are acceptable and compliant with GDC 16, GDC 38, GDC 50 and 10 CFR 52.47(c)(2).

23.T Changes to the Main Control Room Emergency Habitability System

23.T.1 Description of Proposed Changes

In letters dated July 29, 2010, and September 22, 2010, the applicant has proposed changes to the minimum amount of stored compressed air in the MCR VES emergency air storage tanks. To increase the margin to the control room operator dose limits and to expand the site dispersion factors that are permitted for the AP1000 design, a passive filtration subsystem design was added to the MCR VES to filter potential contaminated in-leakage. The subsystem incorporates an eductor, which uses the VES compressed air flow to induce recirculation of MCR air through a filtration unit. The performance of the added subsystem allows for 0.42 cubic meter per minute (m^3/m) (15 cubic feet per minute (cfm) of unfiltered in-leakage while maintaining operator dose below 50 millisievert (mSv) (5 roentgen equivalent man (rem)) total effective dose equivalent (TEDE) required by GDC 19. With the addition of the subsystem, the VES provides a filtration unit to capture potential contaminated air that may leak into the MCR envelope. The applicant conducted testing of the passive filtration subsystem at the Westinghouse Waltz Mill facility. The testing was confirmatory testing to show the performance characteristics of the added passive filtration design and to collect data on the performance of the eductor itself. The passive filter train, utilizing the eductor as well as the technical specifications and ITAAC were reviewed and approved in Sections 6.4 and 9.4.1 of this report. However, after the Chapter 6 and Chapter 9 SERs were issued, the applicant identified the need to adjust the TS on quantity of compressed air needed for 72 hours of continuous operation. The quantity of air needed to support 72 hours of operation increased because the pressure regulating valve minimum required operating inlet pressure needed to increase to ensure that the set outlet pressure could be maintained during the duration of system operation.

The minimum amount of required stored compressed air is changed from 8895 standard cubic meter (m^3) (314,132 standard cubic foot (scf)) to 9276 standard m^3 (327,574 scf. Filling the tanks with 9276 standard m^3 (327,574 scf of air will ensure that the tanks are capable of providing 119 standard cubic meters per hour (scmh) (70 standard cubic foot per minute (scfm)) of air for 72 hours. Operability is determined based on the amount of compressed air stored in the tanks as determined from tank pressure and storage room temperature. The new volume of air is provided in both Tier 1 and Tier 2, including the TS. The relationship between tank pressure, room temperature and volume is provided in the TS Bases.

These proposed design changes also modified the instrumentation representation in DCD Figure 6.4-2 (Sheet 2 of 2) to better represent the instrumentation used in the design. The figure changed a flow instrument from an orifice plate with a differential pressure sensor to a thermal dispersion mass flow transmitter.

23.T.2 Regulatory Basis

The applicable regulations for the control room habitability system aspects of these design changes are detailed in NUREG-0800 Section 6.4, "Control Room Habitability System." For the changes proposed by the applicant, the following are relevant.

- 10 CFR Part 50, Appendix A, GDC 19
- 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS and Bases reflect the VES design and operating information described in DCD

Section 6.4. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance specified in NUREG-0800 Chapter 16.

23.T.3 Evaluation

The proposed design changes were reviewed for compliance with the applicable regulations. This DCP raises the minimum amount of stored compressed air in the VES emergency air storage tanks to ensure the supply of 72 hours of air at the maximum flow rate of 70 scfm. The mission time for the passive safety-related system is 72 hours based on the Commission's policy on passive systems. The design basis requires that 65±5 scfm flow be provided for all DBAs. As a result, there needs to be enough air to provide 72 hours of flow at the maximum flow of 119 scmh (70 scfm).

The change is necessary to accommodate a 1379 kPa (200 psig) pressure regulator minimum inlet pressure instead of 689 kPa (100 psig) prior to the passive filter train. This is done to ensure that downstream pressure and flow are maintained within the required tolerance. The applicant has demonstrated that the new volume of air is adequate to accomplish the design basis functions.

The staff audited APP-VES-M3C-005, "VES Minimum Pressure Calculations," Revision 1. The calculation determines the amount of breathable air required during 72 hours of VES operation. Basic heat transfer equations and air properties were used to determine heat transfer coefficients and transient heat loss from concrete to outside air and the tank package. Cooldown analysis for station blackout during extreme winter weather were used to determine VES tanks pressure decline due to temperature cooldown and air delivery to MCR. Assuming cold conditions is conservative because it minimizes the volumetric flow. The staff has found that the calculations demonstrate there is an adequate amount of gas to maintain pressure above 1379 kPa (200 psig) at the inlet of the pressure regulator. The calculations showed that for the limiting case, there was not much margin. However, because the calculations were done for the limiting cases, small margin is acceptable.

In Chapter 6 of this report, the staff accepted the passive filter train. The staff came to this conclusion, in part, based on there being adequate ITAAC and Technical Specifications to demonstrate that 65±5 scfm flow is provided from the canisters to the eductor and that 600 scfm of control room air would be drawn through the filter train. The COL holder would need to demonstrate prior to plant operation and periodically thereafter that the system would accomplish the design basis functions. Because the passive eductor-driven filter train was new to the nuclear industry, the staff requested that testing be done to prove this design concept was capable of being operated successfully. Prior to the issuance of the Chapter 6 SER, the applicant completed testing to demonstrate that the system was capable of meeting the performance requirements.

When these proposed design changes were submitted by the applicant, the staff wanted to make sure the testing still demonstrated the system was capable of being operated successfully. As such, the staff also audited TS-SEE-111-09-03, "AP1000 VES Air Filtration Test Specification." The testing performed by the applicant had three objectives. The first was to demonstrate 60 scfm is capable of inducing 600 scfm, the second was to determine whether a feed flow rate higher than the design duct flow rate would damage system components, and the third was to demonstrate the system can be operated below maximum allowable noise

levels defined in NUREG-0700, "Human-System Interface Design Review Guidelines." The maximum allowable noise is 65 dB(A).

The staff observed that the testing did demonstrate the system could be built with the necessary performance characteristics. The staff also observed that the test results showed that the combined flow is sensitive to back pressure, feed pressure and feed flow. For example, for the same feed pressure, an increase in backpressure by 0.5 inch of water will reduce combined flow by approximately 170 scmh (100 scfm). Additionally, at the same feed flow an increase in the feed pressure by 34 kPa (5 psi) will reduce combined flow by approximately 170 scmh (100 scfm). As a result, the staff notes that the parameters influencing the induced flow, for example line losses between the regulator and the eductor, will need to be carefully controlled by the COL holder and that the required 1020 scmh (600 scfm) induced flow may not be satisfied at all conditions. The COL holder has the responsibility to ensure that at least 1020 scmh (600 scfm) will be induced by a feed flow rate of at least 1020 scmh (60 scfm).

The two safety-related flow rates will be demonstrated by ITAAC. Additionally, Tier 1 requires the following be demonstrated:

- The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.
- The airflow rate from VES is at least 60 scfm and not more than 70 scfm.
- The system provides a passive recirculation flow of MCR air to maintain main control room dose rates below an acceptable level during VES operation.
- The air flow rate at the outlet of the MCR passive filtration system is at least 600 cfm greater than the flow measured by VES-FT003A/B.
- The noise at the operator station is limited to 65 dB(A).

The staff has concluded that with the ITAAC to demonstrate the capacity of the system design the system will meet the requirements of GDC 19. Additionally, the ITAAC are sufficient to show the as-built plant will function and, as a result, 10 CFR 52.47(b)(1) is satisfied.

With regard to the change in instrumentation, the staff finds that either type of instrument can acceptably measure flow. The instrument is safety-related and subject to the quality assurance requirements. As a result, the staff finds the instrumentation change acceptable as well.

With respect to proposed changes to TS 3.7.6 and its associated bases, the applicant stated that both tank pressure and room temperature are used to determine the acceptable minimum storage capacity of the air tanks in term of standard cubic feet (scf). The acceptance criteria are presented in new Figures B 3.7.6-1 and B 3.7.6-2 for use in the verification of the minimum storage capacity of the air tanks specified in Required Action D.1 and SR 3.7.6.2, respectively. The staff finds these changes acceptable because they reflect the VES design and operating information described in DCD Section 6.4,

23.T.4 Conclusion

The staff finds that there will be an adequate amount of air for the extreme winter conditions for 72 hours of VES operation and to maintain pressure above 1379 kPa (200 psig) at the inlet of the pressure regulator. The staff has concluded that the proposed design change complies with GDC 19 and the acceptance criteria specified in Section 6.4 of NUREG-0800. Lastly, the staff finds that the ITAAC are sufficient to demonstrate that the system when built will accomplish the safety function.

23.U Changes to Main Steam Isolation Valve Subcompartment

23.U.1 Description of Proposed Changes

In letters dated, August 12, 2010, and September 30, 2010, the applicant proposed design changes that make the vent paths associated with the main steam isolation valve (MSIV) subcompartments larger. The applicant also proposed to change the content of the pipe hazards analysis report described in Tier 2 DCD Section 3.6.2.5 and to remove tables in Section 6.2 of the DCD that report mass and energy releases and compartment differential pressures outside of containment.

NUREG-0800 Section 6.2.1.2 describes subcompartment analyses inside of containment. Additionally, NUREG-0800 Section 6.2.1.4 describes mass and energy release from secondary side breaks inside of containment. In Chapter 3 of NUREG-0800, high energy pipe hazards outside of containment are described. In the certified design, subcompartment analyses and mass and energy data for pipe breaks outside of containment are included in Chapter 6. This is not typical and not consistent with NUREG-0800 or RG 1.206, "Combined License Applications for Nuclear Power Plants." Additionally, there is a COL holder item in FSAR Chapter 3 that requires the COL to perform a pipe hazards analysis. As a result, there is some confusion in the certified design. In DCD Section 6.2.1.2 entitled, "Containment Subcompartment Analysis," there are analyses for subcompartments outside of containment. Additionally, in DCD Chapter 3 there is the requirement for another pipe hazards analysis to be performed by the COL holder.

The applicant identified that the rupture of a feedwater pipe may produce more limiting results than the main steam line break that was described in the DCD; therefore, larger vent paths were needed in the main steam valve rooms.

The applicant has proposed to increase the size of the vent paths in the roof of the Auxiliary Building to provide larger vent paths for high-energy hazards. The proposed design changes also modify the structural attachments at the Auxiliary Building roof. The applicant identified a pipe hazard that releases more energy than is currently considered in the hazards analysis. The vent paths are described in Chapter 3. Specifically the applicant has made changes in the doghouse structures on the roof of the Auxiliary Building. Each doghouse structure has a blowout panel and a louvered vent. The blowout panel is 3.04 m (10 ft) by 3.04 m (10 ft) and the louvered vent is 1.82 m (6 ft) by 1.82 m (6 ft).

The applicant has clarified Section 3.6.2.5 to explicitly include subcompartment pressurization of these compartments outside of containment in the pipe hazards analysis report. To eliminate ambiguity in the DCD, the applicant has removed the subcompartment analyses and mass and energy tables for pipe hazards outside of containment from Chapter 6 of the DCD.

23.U.2 Regulatory Basis

For the Chapter 6 changes, the applicable regulations for the containment systems aspects of this design change are detailed in NUREG-0800 Section 6.2.1.2, "Subcompartment Analysis," and include the following:

- 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases"

- 10 CFR Part 50, Appendix A, GDC 50

However, because the subject subcompartments are not inside the containment, GDC 50 is not applicable.

For the Chapter 3 changes, the applicable regulations for the protection against pipe rupture are detailed in NUREG-0800 Sections 3.6.1, "Plant Design For Protection Against Postulated Piping Failures In Fluid Systems Outside Containment"; 3.6.2, "Determination Of Rupture Locations And Dynamic Effects Associated With The Postulated Rupture Of Piping"; and in Branch Technical Position (BTP) 3-3, "Protection Against Postulated Piping Failures In Fluid Systems Outside Containment."

23.U.3 Evaluation

The proposed design changes were reviewed for compliance with GDC 4, which requires that structures withstand the effects of high energy hazards. The design of internal compartments must accommodate the effects of, and be compatible with, the environmental conditions associated with postulated accidents or high energy hazards. The internal compartments shall be appropriately protected against dynamic effects.

The proposed design change of increasing the size of the vents improves the room's or compartment's ability to relieve pressure if a high energy pipe fails in the compartment. Confirmation that the final as-built design is in compliance with the requirements is the responsibility of the COL holder. This responsibility is clearly described in the COL holder item in DCD Section 3.6.4.1. Section 3.6.2.5 of the DCD outlines the content of the pipe hazards analysis report. The applicant proposed to change Section 3.6.2.5 to include the following statement:

> Evaluate compartment pressurization in the break exclusion zones in the vicinity of containment penetrations due to 1.0 square foot breaks in the main steam and feedwater lines.

The proposed change clarifies that the compartment pressurization in the break exclusion zones needs to be evaluated for 1.0 square foot breaks in the main steam and feedwater lines. The proposed design change is acceptable because it improves the ability of the facility to withstand high-energy pipe breaks. The change to Section 3.6.2.5 is acceptable because it clarifies the content of the high-energy line break analysis necessary. A COL holder item for high-energy line break analysis is an appropriate approach to demonstrate compliance with GDC 4 because much of the pipe hazards analysis is site-specific in nature.

The staff also finds the removal of the peak differential pressures and mass and energy tables from DCD Chapter 6.2 acceptable. Section 6.2 is dedicated solely to containment issues.

Neither NUREG-0800 nor RG 1.206 recommends these items be included in Chapter 6. Outside containment subcompartments are much more appropriate in Chapter 3 under the COL holder item. As a result, the staff finds this change acceptable as well.

The DCD Tier 2 Section 3.8.4.3.1.4 specifies that a differential pressure of 41 kPa (6 psid) be the design limit for subcompartment pressurization in the MSIV rooms. The design finalization shows that a main feedwater line pipe rupture without adequate venting will cause the subcompartment pressurization to exceed the 41 kPa (6 psid) design pressure limit. Accordingly, the applicant decided to enlarge the venting area of the roof of the Auxiliary Building in order to meet the requirements of the DCD. The applicant provided the modified steam vent design with the proposed design changes. In addition, the applicant committed to finalize the design of the two doghouse structures on the roof of the Auxiliary Building in accordance with the design procedure of the critical sections as defined in DCD Section 3.8, its appendices, and in technical report (TR)-57, APP-GW-GLR-045, "Nuclear Island: Evaluation of Critical Sections."

During the staff's evaluation of these proposed design changes, the following additional information was requested from the applicant:

- Engineering drawings of the doghouse structure at the roof, detailing the venting assemblies, blow-out panels, louvered vents, and connections to the Auxiliary Building roof.

- Stress evaluation on the structures, including applied design-basis loading and the resulting maximum stress.

- The basis, including acceptance criteria and performance requirements, upon which the proposed new design is determined to be acceptable.

The applicant's September 30, 2010, response stated that the design will be finalized using the same methodology, specifications for load combinations, and safety factors that are specified in DCD Section 3.8. Because the final design will be consistent with the design procedure of the critical sections as defined in DCD Section 3.8, its appendices, and in TR-57, the staff finds the change to the vent area to be acceptable.

23.U.4 Conclusion

On the basis of its review of Sections 3.6.2 and 6.2, the staff finds the proposed design changes acceptable. The approach chosen by the applicant will ensure compliance with GDC 4.

The staff's review concludes that the proposed design changes are acceptable because they will not adversely affect safety-related SSCs or the capability of the MSIV depressurization subcompartments to perform their intended functions of pressure relief for the postulated high-energy line break.

23.V Changes to the Component Cooling Water System

23.V.1 Description of Proposed Changes

A leak or tube rupture in the RCP external heat exchanger (EHX) would not result in over-pressurization of piping outside containment, since the pressure in the CCS is controlled by the system's atmospheric surge tank. Such an event could result in a reactor trip and a nonisolable flow path from the RCS through the CCS piping and the surge tank vent to the turbine building (interfacing system LOCA). The applicant has proposed design changes intended to add a safety-related means to isolate this potential flow path. The proposed design changes would provide automatic, safety-related isolation of a LOCA caused by the rupture of one of the RCP EHX tubes. This automatic isolation would prevent discharge of reactor coolant to the turbine building through the CCS surge tank vent and would limit offsite doses to values below those already found acceptable in the event of a small break LOCA, as stipulated in 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors."

In letters dated July 28, 2010; September 3, 2010; September 29, 2010; and October 18, 2010, the applicant proposed design changes to the CCS. The proposed design changes relate to the following:

- Instrumentation and Controls – Containment isolation:

  The applicant proposed to modify the closure logic for CCS motor-operated containment isolation valves CCS-PL-V200, CCS-PL-V207, and CCS-PL-V208 to add a requirement to close on generation of the RCP bearing water high temperature pump trip signal. This modification would add a new isolation signal to DCD Tier 2, Table 6.2.3-1, "Containment Mechanical Penetration and Isolation Valves."

  A closure signal to the component cooling system containment isolation valves is derived from a coincidence of two of the four divisions of high RCP bearing water temperature for any RCP. The high temperature setpoint and dynamic compensation are the same as used in the high RCP bearing water temperature RCP trip (Section 7.3.1.2.5, Condition 6), but with the inclusion of preset time delay.

- Instrumentation and Controls – CCS:

  The applicant proposed to remove the automatic isolation of the CCS RCP HX outlet isolation valves (CCS-PL-V256A/B/C/D) to close on high deviation between inlet and outlet flows. Simultaneous flow deviations in both the inlet and outlet lines would generate a flow deviation alarm and not isolate these valves. This alarm would be indicative of RCS leak conditions and would alert plant operators to close the valve on the cooling water outlet line on each RCP to prevent reactor coolant flow throughout the CCS. Both the flow signals and the isolation valves are nonsafety-related.

- CCS:

  The applicant proposed to install a 10.16 cm (4 in) x 15.24 cm (6 in) safety-class relief valve, designated CCS-PL-V270 and CCS-PL-V271, respectively, on each of the 25.40-cm (10-in) CCS supply and return lines (total of two relief valves), just inside the

innermost containment isolation valves (CCS-PL-V201 and CCS-PL-V207). In addition, the applicant proposed to change the safety class of the section of line between the innermost containment isolation valves and the Appendix J test valves (CCS-PL-V214 and CCS-PL-V216) from Class '0' to Class 'C' to ensure that the relief valves are installed as ASME safety-class piping.

- Technical Specifications:

  The applicant proposed to add an RCP bearing water temperature high trip function to TS Table 3.3.2-1, "Engineered Safeguards Actuation System Instrumentation," for closure of the CCS containment isolation valves.

These proposed design changes include revisions to the Tier 2 DCD Sections 3.2, 3.9, 3.11, 5.2, 6.2, 7.2, 7.3, 9.2.2, and Chapter 16.

23.V.2 Regulatory Basis

- Instrumentation and Controls – Containment isolation

  The regulatory basis for evaluating the changes to the AP1000 closure logic for CCS motor-operated containment isolation valves is documented in Chapter 7 of NUREG-1793. Review of these proposed design changes is also based on the requirements in 10 CFR 50.55a(h), 10 CFR 52.47, and GDC 20 and GDC 21 of 10 CFR Part 50, Appendix A.

  The applicable regulations for the containment systems aspects of this design change are detailed in NUREG-0800 Section 6.2.4, "Containment Isolation System," and include the following:

  - GDC 16 requires that the containment isolation system allow the normal or emergency passage of fluids through the containment boundary while preserving the capability of the boundary to prevent or limit the escape of fission products from postulated accidents.

  - GDC 54, "Piping Systems Penetrating Containment," requires that the containment isolation system valves in piping systems that penetrate the containment be designed to close reliably under accident conditions and prevent the uncontrolled release of radioactive materials.

- Instrumentation and Controls – CCS:

  The regulatory basis for evaluating the proposed instrumentation and controls changes is documented in Chapter 7 of NUREG-1793. Both the flow signals and the isolation valves for the CCS RCP HX outlet isolation valves are nonsafety-related and are not relied upon to perform any safety functions. However, they are part of the CCS, which is considered to be important to safety because it supports the normal (defense-in-depth) capability of transferring heat from various plant components and also removing reactor system and spent fuel decay heat. Reviews of the changes are based on meeting the relevant requirements of 10 CFR 50.55a(h), 10 CFR 52.47, GDC 13, and GDC 19 of 10 CFR Part 50, Appendix A.

- CCS:

  The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793. While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the RCS is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to RTNSS in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes could not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System – Revision 4, March 2007," as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified in NUREG-0800 Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

  The following regulatory guidance is also applicable to the proposed design changes:

  - RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"

  - RG 1.29

  - SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," related to overpressurization of low-pressure piping systems due to RCS boundary isolation failure could result in rupture of the low-pressure piping outside containment. This could result in a core melt accident with an energetic release outside the containment building, potentially causing a significant offsite radiation release.

- Technical Specifications:

  The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the CCS isolation function design and operating information described in DCD Sections 5.2, 7.3, and 9.2, respectively. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance in NUREG-0800 Chapter 16.

23.V.3 Evaluation

During the staff's evaluation of these proposed design changes, additional information was requested from the applicant. In a letter dated September 3, 2010, the applicant responded to the request for additional information. The staff reviewed the applicant's responses and came to the following conclusions.

- The staff determined that the CCS piping system is adequately protected from over-pressurization due to a postulated RCP external heat exchanger tube rupture with the addition of two new relief valves, one near each CCS containment penetration. Both relief valves are designed to ASME Code Section III, Class 3 and have been added to a markup of Table 3.9-16, "Valve In-service Test Requirements," with a test frequency of at least once per 10 years. The failure of a spring-operated safety valve to open on a high pressure condition is excluded as a single active failure; however, two relief valves will see the over-pressurization event since there is no check valve between the cooling water line serving the reactor coolant drain tank. Since each valve has sufficient capacity to prevent system overpressure for the largest expected discharge from the heat exchanger tube rupture event, the staff finds the CCS piping system is adequately protected from over-pressurization. Therefore, the staff finds the proposed design changes to be acceptable.

- The proposed design changes were reviewed for compliance with GDC 16 and GDC 54 pertaining to containment isolation and containment integrity. The proposed design changes modify the closure logic for the motor operated CCS containment isolation valves CCS-PL-V200, CCS-PL-V207, and CCS-PL-V208 by adding a requirement to close on the generation of a RCP bearing water high temperature signal. This modification adds an additional isolation signal to these valves, and has been included in AP1000 DCD, Tier 2, Table 6.2.3-1, "Containment Mechanical Penetration and Isolation Valves."

  Containment isolation valves are required to close reliably under accident conditions and maintain their integrity to prevent the uncontrolled release of radioactive materials from containment. RCP bearing high water temperature would result from a rupture of one of the U-tubes in the RCP-EHX, which is cooled on the shell side by the CCS. The tube rupture would result in both increased temperature and increased pressure in the CCS. Both GDC 16 and GDC 54 require the containment isolation valves to close to prevent the uncontrolled release of radioactive materials from containment. Therefore, the containment isolation valves must operate at the pressure and temperature conditions in the CCS generated by the RCS tube rupture. Additionally, the containment isolation valves must maintain their integrity and remain leak tight when closed.

  Prior to a rupture, the tube side temperature and pressure in the RCP-EHX are approximately 71 °C (160 °F) and 15.5 MPa (2250 pounds per square inch absolute (psia)); whereas, the temperature and pressure on the shell side of the heat exchanger are approximately 36.6° C (98 °F) and 793 kPa (115 psia). The CCS is a 1379 kPa (200 psig) system, including the containment isolation valves and their included piping. Following a tube rupture the CCS pressure and temperature in the containment would increase significantly. The following conditions must be satisfied:

- The 1379 kPa (200 psig) design containment isolation valves CCS PL-V207, PL-V208, PL-V200, and PL–V201 must be able to close against the increased pressure due to the RCP-EHX tube rupture.

- The 1379 kPa (200 psig) design containment isolation check valve CCS-PL-V201 must maintain its integrity at the increased pressure and temperature due to the RCS-EHX tube rupture and remain closed.

- The new safety-class relief valves, CCS-PL-V270 and CCS-PL-V271, must operate under the conditions of flashing fluid at the maximum fluid temperature produced by the RCP-EHX tube rupture event.

The applicant proposed to add a single safety-class relief valve on each of the 25.4 cm (10 in) CCS supply and return lines respectively just inside the innermost containment isolation valves, CCS-PL-V201 and CCS-PL-V207. The safety-class relief valves, with a set pressure of 1379 kPa (200 psig), are intended to limit the CCS pressure at the containment penetrations to the design conditions for a 1379 kPa (200 psig) system.

The applicant performed an analysis, AP1000/ANSALDO RELAP calculation APP-CCS-M3C-164, Revision 0, September 24, 2010, to simulate the occurrence of a postulated double ended tube break in the RCP-EHX in order to evaluate CCS transient and steady state conditions of temperature and pressure at the relief and containment isolation valves.

The results of the analysis for Case 1, which postulates both safety class relief valves providing overpressure protection and opening at the set pressure of 1379 kPa (200 psig), demonstrate the following:

- CCS-PL-V201 reaches a maximum pressure of 1482 kPa (215 psia), a maximum temperature of approximately 193.3 °C (380 °F), and reaches a steady state condition of approximately 1241 kPa (180 psia) and 189 °C (372 °F) at 1000 seconds.

- CCS-PL-V207 reaches a maximum pressure of 1489 kPa (216 psia), a maximum temperature of approximately 193.3 °C (380 °F), and reaches a steady state condition of approximately 1241 kPa (180 psia) and 189.4 °C (373 °F) at 1000 seconds.

The results of the analysis for Case 2, which postulates a single relief valve, V270, opening upon reaching the set pressure of 1480 kPa (200 psig), and a single failure of relief valve V271 to open, demonstrate similar results of temperature and pressure, but reach steady state conditions at 400 seconds.

The AP1000 piping class sheets and standard details, APP-PLO2-Z0-001, Revision 5, p. 120, provide the design pressure and temperature envelope for class JCB and JCC piping and the CCS containment isolation and safety-related relief valves, summarized as follows;

0-37.7 °C (32-100 °F)	65.5 °C (150 °F)	148.8 °C (300 °F)
1965 kPa	1862 kPa	1586 kPa
285 psig	270 psig	230 psig

Except for the CCS temperature of ~193.3 °C (~380 °F), the transient and steady state conditions are well within the pressure and temperature design envelope of the containment isolation valves and piping. In a subsequent revision to the valve and piping data sheets, "AP1000 Piping Class Sheets and Standard Details", APP-PLO2-Z0-001, Revision 5, the applicant made appropriate changes which resolves this issue.

A RELAP calculation by staff confirmed the above transient results from the applicant's analysis, which was submitted in a letter dated October 18, 2010. The staff concludes that the containment isolation valves would be able to close at these pressure and temperature conditions, and maintain their integrity following a tube rupture in the RCP-EHX.

Each of the four CCS containment isolation valves will be explicitly identified in DCD Tier 2, Table 3.9-12 (Sheet 1 of 7) as having both containment isolation and accident mitigation functions. The new safety-related relief valves, V270 and V271, will be explicitly identified as having an accident mitigation function. The conditions applicable for accident mitigation will be added to the valve data sheets as notes pertaining to the specific fluid temperature and pressure conditions for which the valves must remain operable and intact. In a subsequent revision to the valve and piping data sheets, "AP1000 Piping Class Sheets and Standard Details", APP-PLO2-Z0-001, Revision 5, the applicant made appropriate changes which resolves this issue.

RG 1.26 recommends that portions of cooling water systems important to safety and which are designed for the functioning of components important to safety be designed to Quality Group C standards. For piping and valves the applicable standard is ASME Section III, Class 3.

The applicant proposed to install two additional relief valves, CCS-PL-V270 and CCS-PL-V271, one on each of the 25.4-cm (10-in) CCS supply and return lines, just inside the innermost containment isolation valves, CCS-PL-V201 and CCS-PL-V207, respectively. These relief valves are intended to limit the CCS pressure at the containment penetrations to approximately 1379 kPa (200 psig). Since these relief valves are provided to protect the safety-related CCS containment isolation valves and piping, which are designed to Quality Group B standards, they will be designed to Quality Group C criteria, or ASME Code Section III, Class 3. The staff has evaluated this designation, which will be added to the DCD Tier 2, Table 3.2-3, "AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment," and concludes that it meets the recommendations of RG 1.26.

SSCs of a nuclear power plant designated as seismic Category I and designed to withstand the effects of an SSE and remain functional include components affecting the safety-related function of the primary containment, in accordance with RG 1.29. The relief valves, CCS-PL-V270 and -V271, are provided to protect the operability and long-term integrity of the CCS 25.4-cm (10-in) containment penetrations. Therefore, they will be designed to seismic Category I criteria. The staff has evaluated this designation and concurs that it meets the guidance of RG 1.29. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

For advanced reactor design, the staff stated its position regarding intersystem LOCA (ISLOCA) protection in SECY-90-016. The staff stated that advanced light-water reactor (ALWR) designs should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to full RCS pressure.

The CCS is a low-pressure system design with an ultimate rupture strength (URS) below RCS operating pressure. Overpressurization of the CCS could occur following a tube rupture in the RCP-EHX. A relief valve has been added to each of the CCS supply and return headers inside the containment isolation valves in containment to prevent overpressurizing the CCS. The adequacy of pressure relief in protecting the operability and integrity of the containment isolation of CCS has been addressed above and found acceptable. Therefore, the staff finds CCS overpressure protection by the two safety-related relief valves meets the intention of the guidance for ISLOCA.

- Since DCD Tier 1 testing of the CCS-PL-V200 and V201 valves is adequately described in Table 2.2.1-1, the staff determined no other Tier 1 changes or ITAAC are required. DCD Tier 2, Section 14.2.9.2.5 adequately addresses CCS testing, which includes proper operations of controls, instrumentation, actuation signals and interlocks. Furthermore, DCD Tier 2, Section 14.2.9.1.10, "Containment Isolation and Leak Rate Testing," addresses testing for proper operation of safety-related containment isolation valves listed in Table 6.2.3-1. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

  The proposed revision to DCD Section 9.2.2.4.5.2 provides an adequate description of the automatic isolation function. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

  The applicant proposed to modify the closure logic for CCS motor-operated containment isolation valves CCS-PL-V200, CCS-PL-V207, and CCS-PL-V208 to add a requirement to close on generation of the RCP bearing water high temperature pump trip signal. If a tube break occurs at normal (higher) RCS temperatures, the RTDs used for the automatic isolation function sense the temperature of the reactor coolant flowing through the collection header almost immediately after the break initiates. However, for an idled RCP with the RCS at low temperatures, such as near 93 °C (200 °F), RCS flow is much less (as low as 5 percent) than the flow rates seen at 100percent shaft speed. Even with reactor coolant temperature near 93 °C (200 °F), the RTDs will still sense the reactor coolant temperature within minutes and produce the automatic RCP bearing water high temperature trip, which closes the CCS containment isolation valves. Therefore, the staff's review finds the applicant's design change to using reactor coolant temperature instruments to initiate CCS containment isolation acceptable.

- The change to add Modes 3 and 4 to Table 3.3.2-1 of the TS is acceptable to the staff since it envelops the possible Modes within which this event is postulated to occur. The staff had additional questions concerning how the applicant would affect this trip in Modes 3 and 4 since the temperature differential between the RCPs and CCS may be so small that a trip signal would be delayed longer than necessary to effectively isolate the system. In its response, the applicant provided further details of the sensor arrangement and flow path of coolant for this event, should it occur in Modes 3 or 4. The staff found this discussion acceptable because it describes how the trip signal will be generated in sufficient time to isolate the system, without the need for permissive or

interlocked trip signals. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- The applicant proposed to add a signal to Functional Diagram 7.2-1 (Sheet 5 of 20) in AP1000 DCD Tier 2 to close the component cooling system containment isolation valves. This closing signal is derived from the existing 2-out-of-4 control logic with bypass capacity of the high RCP bearing water temperature for any RCP. The high temperature setpoint and dynamic compensation for this new closing signal are the same as those used for RCP trips based on the high RCP bearing water temperature, but it will be implemented with a preset time delay. The newly added logic provides safety-related protection system functions and also meets the reliability and test requirements for the safety protection system as required by GDC 20 and GDC 21. Hence, the staff concludes that this change is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

  The applicant has also proposed to remove the automatic control function to isolate the CCS RCP HX outlet isolation valves on a high delta-flow between the HX inlet and outlet flow. The high delta-flow between the HX inlet and outlet cooling water lines now will only generate a flow deviation alarm in the MCR. The original automatic isolation function is provided by the plant control system (PLS) and non-Class 1E sensors and instrumentation. This nonsafety-related isolation control function is redundant to and is replaced by the automatic closure of the CCS containment isolation valves based on the high RCP bearing water temperature produced through the safety-related PMS using safety-related RCP RTD sensors. In addition, a new alarm for the high delta-flow between the CCS RCP HX inlet and outlet cooling water lines is now provided in the MCR, and also the remote manual operation of CCS RCP HX outlet isolation valves is retained. If annunciated in the MCR, the new alarm would alert plant operators to close the RCP HX outlet isolation valve remotely in the MCR on the cooling water outlet line to prevent reactor coolant flow throughout the CCS. The staff finds that the above changes meet relevant criteria as required in 10 CFR 50.55a(h) and 10 CFR Part 50, Appendix A GDC 13 and GDC 19. The staff, therefore, concludes that these changes are acceptable.

23.V.4 Conclusion

The staff's review concludes that these proposed design changes are acceptable because they will not adversely affect safety-related SSCs; and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. Adequate overpressure protection of the CCS is provided with ASME Code relief valves during a postulated RCP external heat exchanger tube rupture. An automatic signal will be generated to the associated containment isolation valves in the event of a postulated RCP external heat exchanger tube rupture, as shown in TS Table 3.3.2-1.

On the basis of its review of the containment isolation design aspect of the proposed CCS overpressure protection design change, the staff concludes that the design complies with the acceptance criteria in Section 6.2.4 of NUREG-0800, including: GDC 16; GDC 54; RG 1.26; and RG 1.29.

The staff also finds the design change to provide CCS overpressure protection by the two safety related relief valves meets the intention of the guidance for ISLOCA found in SECY-90-016.

23.W Changes to Add a Vacuum Relief System to the Containment

23.W.1 Description of Proposed Changes

In letters dated August 16, 2010; September 29, 2010; and October 15, 2010, the applicant submitted proposed design changes and supporting documentation, which add a containment vacuum relief system to the existing containment air filtration system (VFS) vent line penetration. The NRC staff reviewed the system design, containment external pressure analyses, containment isolation functions, leak rate testing, descriptions of the valve design, qualification and IST programs, I&C, and associated TS.

The applicant's proposed design changes add a vacuum relief system to the existing VFS 40.64-cm (16-in) vent line penetration as seen in DCD Tier 2, Figure 9.4.7-1, "Containment Air Filtration System P&ID", Sheet 1 of 2. The proposed vacuum relief system consists of redundant vacuum relief devices sized to prevent differential pressure between containment and the shield building from exceeding the design value. Each of the two vacuum relief device flow paths consists of a check valve (VFS-PL-V803A/B) inside containment; a motor operated butterfly valve (VFS-PL-V800A/B) outside containment; and associated piping. Each of these four valves also has a containment isolation function. The redundant check valves inside containment share a common inlet line with the vent line penetration and have independent discharge lines into containment. The redundant motor operated butterfly valves outside containment share a common inlet line. Each relief device, consisting of a check valve, connecting piping, and a motor operated valve (MOV), is designed to provide 100 percent of the required capacity to prevent a differential pressure across the containment vessel from exceeding the design value. Each relief flow path provides the required capacity, such that a single failure of any of the relief devices would not limit the flow below what is required to mitigate a containment vacuum relief event.

The normally closed butterfly valves are designed with motor operators that are powered from separate Class 1E direct current (dc) battery sources. They are designed to close within 30 seconds of receipt of either an automatic containment isolation signal, or a manual isolation signal. They are designed to open automatically within 30 seconds when the containment pressure signal reaches Low-2 level and remain open to preclude exceeding the containment external design pressure. While the vacuum relief system MOVs are open, the containment will be at a vacuum and flow will be into containment. Once the vacuum condition inside containment is reduced to near ambient pressure conditions, the open signal would automatically clear. This would allow the vacuum relief system MOVs, VFS-PL-V800A/B, to close automatically in the event that a containment isolation signal or high radiation signal is present. The check valves are balanced to remain closed during normal operations, including containment vessel venting.

The proposed vacuum relief system is being added to the existing VFS. A short description of the vacuum relief system is being added to DCD Section 9.4.7. Additionally, the proposed vacuum relief system design is shown on DCD Tier 2, Figure 9.4.7-1, Sheet 1 of 2, "Containment Air Filtration System P&ID." Since the proposed vacuum relief system valves are also part of the containment isolation design, valves VFS-PL-V800A/B and VFS-PL-V803A/B have been added to DCD Tier 2, Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves," with two notes. The first, Note 8, indicates that the valves VFS-PL-V800A/B close on either a containment isolation signal or a high radiation in containment signal, and

open on a Low-2 containment pressure signal. The second, Note 9, indicates that the Type C leak rate testing of the valves VFS-PL-V800A/B would be in the reverse direction and that closure would occur in 30 seconds.

The applicant has changed the containment external design pressure specified in DCD Section 3.8.2.1.1 from 20.0 kPa (2.9 pounds per square inch differential (psid)) to 11.7 kPa (1.7 psid) based on the actuation point of the vacuum relief system. The applicant has also changed the containment external design analysis in DCD Section 6.2.1.1.4 to demonstrate the vacuum relief system is sufficient to mitigate the maximum expected external pressure.

The applicant has also modified AP1000 DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirements," to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. The IST table will specify VFS-PL-V800A/B as motor-operated butterfly valves with safety-related missions to maintain close, transfer close, maintain open, and transfer open; safety functions as active, containment isolation, safety seat leakage, and remote position; ASME Code Class 2 and IST Category A valves; and IST type and frequency as remote position indication and exercise every 2 years, containment isolation leak test, exercise full stroke quarterly and operability test. The IST table will specify check valves VFS-PL-V803A/B as relief valves with safety-related missions as maintain close, transfer close, and transfer open; safety functions as active, containment isolation, and safety seat leakage; ASME Code Class 2 and IST Category AC valves; and IST type and frequency as containment isolation leak test, exercise full stroke every refueling shutdown, and vacuum relief test every 2 years.

The applicant's proposed design changes include the addition of TS 3.6.10, "Vacuum Relief Valves," which address the proposed containment vacuum relief valve operation. The proposed design changes add a function to Table 3.3.2-1, "Engineered Safeguards Actuation System Instrumentation," for the signal which would open the containment vacuum relief valves on a Low-2 containment pressure level. In addition, the proposed changes extend the Mode Applicability for TS 3.6.4, "Containment Pressure," and TS 3.6.5, "Containment Air Temperature," to include Modes 5 and 6, which support TS 3.6.10 requirements.

The applicant has also proposed corresponding changes to Tier 1 DCD sections.

23.W.2 Regulatory Basis

The regulatory basis for evaluating the applicant's proposed design changes and supporting documentation is documented in Chapters 3, 6, 7, 9, and 16 of NUREG-1793.

The applicable regulations for the design, analyses, containment isolation, and containment leak rate testing aspects of the applicant's proposed design changes are detailed in NUREG-0800 Sections 6.2.1.1.A, "PWR Dry Containments"; 6.2.1, "Containment Functional Design"; 6.2.4, "Containment Isolation System"; 6.2.6, "Containment Leakage Testing"; and 9.4.3, "Auxiliary and Radwaste Area Ventilation System." Specifically, the following regulatory requirements and guidance apply:

- 10 CFR Part 50, Appendix A, GDC 2
- 10 CFR Part 50, Appendix A, GDC 16
- 10 CFR Part 50, Appendix A, GDC 38

- 10 CFR Part 50, Appendix A, GDC 50
- 10 CFR Part 50, Appendix A, GDC 52
- 10 CFR Part 50, Appendix A, GDC 53, "Provisions for Containment Testing and Inspection"
- 10 CFR Part 50, Appendix A, GDC 54
- 10 CFR Part 50, Appendix A, GDC 56
- 10 CFR Part 50, Appendix J
- 10 CFR 50.34(f), "Contents of applications; technical information," regarding Three Mile Island (TMI) Action Plan Item II.E4.2, "Containment Isolation Dependability"
- 10 CFR 52.47(b)(1), which requires that a DC application include the appropriate ITAAC
- RG 1.26
- RG 1.141, "Containment Isolation Provisions for Fluid Systems"
- BTP 6-4, "Containment Purging During Normal Plant Operations."

The applicable regulations for the staff's review of the functional design, qualification, and IST programs for the valves described in these proposed design changes are detailed in NUREG-0800 Section 3.9.6 (Revision 3), "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." The staff also considered guidance provided in applicable Commission SECY papers, Commission SRM, GLs, RGs, and regulatory issue summaries. Specifically, the following regulatory requirements apply:

- 10 CFR Part 50, Appendix A, GDC 1
- 10 CFR Part 50, Appendix A, GDC 2
- 10 CFR Part 50, Appendix A, GDC 4
- 10 CFR Part 50, Appendix A, GDC 54
- 10 CFR 50.55a(f), which requires that valves whose function is required for safety be assessed for operational readiness in accordance with the applicable revision to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code)
- 10 CFR 50.55a(b)(3), which takes exception to, or supplements, the ASME OM Code provisions for components within the scope of the IST Program

The applicable regulations for the I&C aspects of the applicant's proposed design changes are detailed in NUREG-0800 Section 7.3, "Engineered Safety Features," and NUREG-0800

Section 7.5, "Information Systems Important to Safety." Specifically, the following regulatory requirements and guidance apply:

- 10 CFR Part 50, Appendix A, GDC 13
- 10 CFR Part 50, Appendix A, GDC 19
- 10 CFR Part 50, Appendix A, GDC 20
- 10 CFR Part 50, Appendix A, GDC 21
- 10 CFR 50.55a(h), which requires that protection systems must meet the requirements stated in either IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in IEEE Standard 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995
- RG 1.97

The applicable regulations for evaluating the applicant's proposed TS are detailed in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." The staff's evaluation focused primarily on confirming that changes to the GTS reflect the containment systems design and operating information in DCD Sections 7.3 and 9.4.7, and the containment analyses described in DCD Section 6.2, and that changes to the GTS meet the requirements of 10 CFR 50.36, "Technical specifications."

23.W.3 Evaluation

23.W.3.1 System Design and Analyses

The staff reviewed the proposed design changes for compliance with GDC 16, which requires that the reactor containment and associated systems be designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The proposed vacuum relief system includes an interlock with the inboard containment isolation valve for the containment purge line. The interlock prevents the vacuum relief system from opening if the inboard isolation valve is open. The staff requested that the applicant explain how the safety function would be accomplished if the inboard valve was open and the interlock prevented the vacuum relief system from functioning. The applicant responded that the vacuum relief system actuation had priority over the interlock and that if a containment vacuum condition existed, the system would actuate regardless of the interlock. The applicant revised the statements in the DCD to clarify this design feature. Specifically, the new DCD Section 7.6.2.4 states that if a vacuum relief actuation signal is present, the vacuum relief signal takes precedence over the valve closure interlock. This design feature ensures the safety function is accomplished and removes the possibility of a single failure associated with the interlock disabling the system.

The staff also reviewed the revised external pressure analysis in DCD Section 6.2.1 for compliance with GDC 16. Conformance with acceptance criteria from NUREG-0800 Section 6.2.1.1.A formed the basis for concluding whether GDC 16 was satisfied.

In DCD Revision 15, the applicant determined that the worst case event for maximum external pressure was the loss of all ac power sources during extreme cold weather. The analysis was conducted using the WGOTHIC code with the following conservatisms:

- External temperature boundary condition set at the minimum site parameter from DCD Tier 1, Table 5.0-1, -40 °C (-40 °F).
- Initial containment relative humidity set to 100 percent to maximize the vapor content and allow for the greatest reduction in pressure due to steam condensation.
- Loss of ac power event immediately reduces the containment heat load to zero.
- Initial containment temperature set at the maximum temperature LCO from TS 3.6.5, 48.8 °C (120 °F).
- Initial containment pressure set at the minimum pressure LCO from TS 3.6.4, -1.38 kPa (-0.2 psid).
- The velocity of air flowing over the containment shell is set at 7.56 meters per second (m/s) 24.8 feet per second (ft/s), which correlates to an external wind speed of 77.24 kilometers per hour (km/h) (48 miles per hour (mph)).
- No air leaks into containment.

The analysis demonstrated the differential pressure across the containment vessel remained below the design value for one hour, which was found to be sufficient time for plant operators to take action to mitigate the event.

The applicant revised the external pressure evaluation to incorporate the vacuum relief system and eliminate the need for operator action, to remove unnecessary conservatisms associated with the first three bulleted items above (no changes were made to the final four assumptions), and to incorporate design changes associated with the DCD amendment. The analysis was based on the WGOTHIC evaluation model approved by the staff in Section 23.S of this report, modified for use in an external pressure transient. The worst case event remains the loss of all ac power during cold weather. The applicant recognized that an operating reactor does not produce enough heat to raise the internal temperature to the maximum value on a cold day; thus, the combination of the maximum internal temperature and minimum external temperature is non-mechanistic. In order to remove conservatism, the first step in the revised analysis is determining the minimum external temperature that can sustain the internal temperature at 48.8 °C (120 °F). This pre-transient stage of the analysis assumed a heat rate equal to the value used to size the active containment cooling system. In order to minimize the external shell heat transfer coefficient, the annulus air flow was set to natural convection. The applicant used this model to determine equilibrium containment temperatures associated with various external temperatures and found that -3.88 °C (25 °F) is the minimum external temperature capable of maintaining the maximum internal temperature (48.8 °C (120 °F)).

The next step in the analysis is the loss of power transient. In this phase, the initial temperatures of the internal containment volume, containment shell, baffle and shield building were set to the equilibrium values found during the pre-transient run. For the design basis run,

the internal temperature is 48.8 °C (120 °F) and the external temperature is -3.88 °C (25 °F), which bounds the proposed LCO for TS 3.6.10 requiring the internal/external temperature differential be less than or equal to 32.2 °C (90 °F). Because the initial conditions exceed the TS LCO, the equilibrium temperatures derived for the shell, shield building and baffle were also found to be bounding.

The original analysis assumed zero heat loss during the transient. This is non-mechanistic because it neglects the contribution of reactor system sensible and decay heat. In order to remove some of this conservatism, the heat load to containment is assumed equal to the sensible heat from a reactor in Mode 3 at normal operating temperature and normal operating pressure. Decay heat is not considered because it is assumed the reactor has never been critical. The staff agrees that these assumptions produce conservatively low heat loads. The applicant stated that the previous assumption of 100 percent relative humidity was also non-mechanistic because the temperature of the containment vessel is below the dew point. As a result, the revised analysis uses a relative humidity of 82 percent, representing a 25 percent margin over the equilibrium value. The staff finds this approach realistically bounds the relative humidity.

During the transient, the external temperature is assumed to decrease -1.11 °C (30 °F) per hour from an initial value of -3.88 °C (25 °F) until it reaches the minimum site value of -40 °C (-40 °F) at 7800 seconds. To demonstrate this assumption is bounding, The applicant evaluated hourly meteorological data gathered at Charlotte, North Carolina from January 1, 1975 to June 24, 1996 and at Duluth, Minnesota between January 1, 1975 and January 1, 2010. Charlotte, North Carolina was chosen by the applicant as a location having typical meteorological behavior for the Southeast Regional Climate Zone. The applicant also chose the Duluth, Minnesota location because it represents the basis for the AP1000 DCD minimum allowable operation temperature of -40 °C (-40 °F). The applicant found that the maximum observed hourly temperature decrease in Charlotte, North Carolina was -6.66 °C (20 °F) (from 22.77 °C (73 °F) to 11.66 °C (53 °F)). The applicant also found that the maximum observed hourly temperature decrease in Duluth, Minnesota, during below-freezing conditions was -8.33 °C (17 °F) (from -7.22 °C (19 °F) to -16.66 °C (2 °F)). The staff finds the applicant's use of a -1.11 °C (30 °F) per hour temperature drop to be reasonable because the applicant's analysis included several recent years worth of data at a typical southeast regional site (Charlotte, North Carolina) and a typical cold weather site (Duluth, Minnesota), and because there is significant margin between the observed values (-6.66 °C (20 °F) for Charlotte, North Carolina and -8.33 °C (17 °F) for Duluth, Minnesota) and the assumed hourly decrease (-1.11 °C (30 °F).

The WGOTHIC model incorporated one 15.2-cm (6-in) valve with a conservatively calculated system resistance, designed to open 20 seconds after the internal containment pressure reached the setpoint of -8.3 kPa (-1.2 psid). The safety analysis limit -8.3 kPa (-1.2 psid) is clearly identified in the DCD and will be used to develop the Low-2 containment pressure setpoint under the TS setpoint methodology program. The 20 seconds required to develop full flow is consistent with the mechanical design requirements identified in Section 9.1.1 of Enclosure #4 of the applicant's October 15, 2010 submittal. Because it is reasonable to expect a butterfly valve to allow significant flow at 60 percent of the stroke, this is also consistent with the proposed acceptance criteria for ITAAC Item 2 from Table 2.7.6-2 requiring the valves to open within 30 seconds.

The transient response demonstrated that the vacuum relief system limits the containment pressure to a minimum value of -11.2 kPa (-1.63 psid), which is bounded by the design value of

-11.7 kPa (-1.7 psid). As described in Enclosure #6 of its October 15, 2010, submittal, the applicant ran additional vacuum relief scenarios at external temperatures of -40 °C (-40 °F) and 10.0 °C (50 °F) and used the results to confirm that the design basis case was limiting. The -40 °C (-40 °F) case, initiated with an internal temperature of 31.11 °C (88 °F), was also used to demonstrate that the internal to external temperature differential LCO is not required if the internal containment temperature is less than 31.11 °C (88 °F). This bounds TS 3.6.10 Required Action B.2 to reduce containment average temperature to ≤ 26.66 °C (≤ 80 °F) when the inside to outside differential air temperature does not meet the LCO.

The staff reviewed the analysis basis during August 27, 2010, and October 3, 2010, audits of APP-SSAR-GSC-112, "AP1000 External Pressure Analysis to Confirm Sizing of the Vacuum Relief System," Revision 0 and Revision 1. This report states that while various transients were considered for the external pressure evaluation, the loss of ac power event remains the most limiting. This is the same event that was found to be bounding in the original analysis, for the reasons discussed in Section 6.2.1 of NUREG-1793. The staff finds this evaluation remains applicable to the revised design because the proposed design changes will not impact determination of the limiting event.

The analysis applied a 7.56 m/s (24.8 ft/s) annulus velocity during the transient to represent an external wind of 77.24 km/h (48 mph). The staff finds this will produce conservatively high heat transfer coefficients because the basis for the correlation was testing on a prior version of the shield building. As discussed in Section 23.S, there is an increased pressure drop associated with the revised shield building, so the annulus velocity associated with a 77.24 km/h (48 mph) wind speed will be less than 7.56 m/s (24.8 ft/s).

The staff conducted an additional audit on August 30, 2010, on three supporting calculation notes. The staff found the baseline WGOTHIC external pressure model, described in APP-SSAR-GSC-746, Revision 0, acceptable because it used the same methodology as the DCD Revision 15 analysis to transform the LOCA model into an external pressure model. The staff reviewed the calculations for the total system resistance in APP-VFS-M3C-224, "Containment Vacuum Relief System Resistance Calculation," Revision 0. The analysis was found to be conservative because the resulting value represents the resistance associated with the more limiting single flow path configuration (which assumes two failures) rather than the worst case single failure. It also incorporates a 65 percent margin over the calculated value to account for potential design changes. The staff reviewed the design basis for the minimum heat load into containment as described in APP-SSAR-GSC-003, "Calculation of the Total Loss of RCS Heat from Mode 3 with Station Blackout," Revision 0. The staff finds this analysis to be conservative because the code used to generate the heat load does not consider contributions from the major component support structures, which are typically large contributors to containment sensible heat.

The staff performed a confirmatory analysis using the CONTAIN computer code and the AP1000 model described in Section 23.S.3 of this report. The model was altered to incorporate one vacuum relief line and to remove the shield building in order to apply the annulus air velocity directly to the external containment shell, producing conservatively high heat transfer coefficients on this surface. The model incorporated the same valve parameters as the applicant except the valve opening delay time was increased to 30 seconds. The staff evaluated the design basis case and the two sensitivity studies. The containment internal pressure transients demonstrated the design basis case, with a minimum negative pressure of -10.8 kPa (-1.56 psid), was limiting. These results are consistent with the applicant's evaluation.

NUREG-0800 Section 6.2.1.1.A recommends that the margin between the external design pressure and the conservatively calculated minimum pressure be at least 10 percent at the construction permit (CP) or DC stage of review. The requirement at the operating license stage of review is that the maximum calculated value be less than the design value (zero-margin criterion). The more restrictive margin is applied at the CP and DC stage to account for revised or upgraded analytical models or minor changes that may result in a decrease in the margin in the as-built design. The margin in the revised design basis analysis is 4 percent. The applicant concluded that this margin is acceptable because the calculation of the vacuum relief system resistance included a 65 percent margin to accommodate system variances associated with plant construction. The applicant claims this addresses the "as-built" considerations that form the basis for NUREG-0800 recommended 10 percent margin at the DC stage. Furthermore, when the margin was removed from the loss coefficient, the resultant minimum external pressure was 10.1 kPa (1.46 psid), which provides a 14 percentmargin to the design value. The staff agrees this is an acceptable approach because it meets the intent of the NUREG-0800 recommendation. This finding is consistent with the staff evaluation in NUREG-1793 Section 6.2.1 where a zero-margin criterion was deemed acceptable for the AP1000 peak accident pressures at the DC stage.

The staff reviewed the proposed containment vacuum relief design for compliance with the requirements of GDC 2, which requires that the system be capable of withstanding the effects of earthquakes. The new portions of the system are safety-related and seismic Category I and, therefore, meet the requirements of GDC 2. However, in Section 9.4.7 of the DCD, the applicant removed statements that demonstrate the nonsafety-related portions of the VFS system meet GDC 2. The staff requested that the applicant explain how the VFS system meets GDC 2. The applicant revised the discussion in DCD Section 9.4.7 to state that system equipment and ductwork whose failure could affect the operability of the safety-related systems or components are designed to seismic Category II requirements. This conforms to the certified DCD Revision 15 design, and is adequate to demonstrate compliance with GDC 2.

As a result, the staff finds that the design and design features described in the DCD ensure the requirements of GDC 2 and GDC 16 are met. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. The staff's evaluation of the AP1000 steel containment, including load combinations and design and analysis procedures, is described in Section 3.8.2 of this report.

23.W.3.2 Containment Isolation and Leak Rate Testing

The staff reviewed the proposed vacuum relief system design for compliance with GDC 56, which requires that each line that connects directly to the containment atmosphere and penetrates primary reactor containment be provided with containment isolation valves. One acceptable design includes: one automatic isolation valve inside, and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. The design of the vacuum relief system complies with the requirements of GDC 56.

10 CFR 50.34(f)(2)(xiv)(B) requires that each non-essential penetration (except instrument lines) shall have two isolation barriers in series. The vacuum relief design consists of two lines, which connect directly with the containment atmosphere and penetrate the primary containment. This design complies with the requirement of GDC 56 by providing each vacuum relief device with a check valve inside containment and a motor operated butterfly valve outside containment. This design complies with the 10 CFR 50.34(f)(2)(xiv)(B) redundancy requirement. If a check

valve failed to close during an accident, the MOVs in series with it would close on a "T" signal (described in DCD Section 6.2.3), thereby providing containment isolation.

To meet the requirements of GDC 56, upon loss of actuating power, automatic isolation valves should take the position of greatest safety. All power-operated isolation valves should have position indications in the MCR. The safe post-accident position for the valves in the vacuum relief system is closed. The motor operated butterfly valves would close on a "T" containment isolation signal. To improve the reliability of the isolation function, these valves also close on a high radiation in containment signal. These valves are each powered by Class 1E batteries to ensure that they would close on these signals. Their position indication in the MCR is shown on DCD Table 6.2.3-1. Valves VFS-PL-V803A and VFS-PL-V803B are self actuated check valves, which would close if a vacuum does not exist inside containment. These check valves would either close or remain closed post-accident. Since the proposed vacuum relief system valves are also part of the containment isolation design, VFS-PL-V800A/B and VFS-PL-V803A/B have been added to DCD Tier 2, Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves," with two notes. The first, Note 8, indicates that the valves VFS-PL-V800A/B close on either a containment isolation signal or a high radiation in containment signal, and open on a Low-2 containment pressure signal. The second, Note 9, indicates that the Type C leak rate testing of the valves VFS-PL-V800A/B would be in the reverse direction and that closure would occur in 30 seconds. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff also reviewed these proposed design changes for compliance with the requirements of GDC 54, as it relates to providing piping systems penetrating the containment with containment isolation capabilities having redundancy and reliability which reflect their importance to safety. To meet the reliability requirements, the components performing a containment isolation function are acceptable if Group B quality standards, as defined in RG 1.26, apply, and the components are designated seismic Category I in accordance with RG 1.29.

The containment isolation section of the vacuum relief design, consisting of the containment isolation valves and the included piping, are designed to ASME Code Section III, Class 2 criteria. The containment penetrations are classified as Quality Group B, as defined in RG 1.26, and seismic Category I. These designations are shown in DCD Tier 2, Figure 9.4.7-1 and Table 6.2.3-1, and Tier 1, Table 2.2.1-1, Table 2.2.1-2, and Figure 2.2.1-1, "Containment Isolation System." The applicant has selected the appropriate mechanical design classification.

10 CFR 52.47(b)(1) requires that a DC application include the appropriate ITAAC. The ITAAC for the containment isolation MOV closing time are shown in DCD Tier 1, Table 2.7.6-2 and Table 2.2.1-3. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In meeting the requirements of GDC 54, relating to lines which provide open paths from the containment to the environs, such as containment purge and vent, the closure times of the isolation valves should minimize the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and to prevent degradation of emergency core cooling system effectiveness by reduced containment back-pressure. CSB BTP 6-4 provides additional guidance on the design and use of the containment purge systems, which may be used during the normal plant operating modes.

The VFS is used to purge the containment atmosphere of airborne radioactivity during normal plant operation. The proposed containment vacuum relief system is a safety grade system used

to mitigate a containment external pressure scenario, and is part of the VFS, sharing the same containment penetration. The purge system is designed in accordance with CSB BTP 6-4. The purge component of the VFS uses 40.6-cm (16-in) supply and exhaust lines and containment isolation valves designed to close in 10 seconds. The vacuum relief component of the VFS uses 15.2-cm (6-in) supply lines and containment isolation valves designed to close in 30 seconds.

In the event of a LOCA, a maximum time of 30 seconds for closure of the 40.6-cm (16-in) valves was assumed for the analysis for the radiological consequences, as seen in DCD Tier 2, Table 15.6.5-2. This closure time is conservative, as the valve design closure time is 10 seconds. This closure time is consistent with the guidance in CSB BTP 6-4, and was found acceptable as an assumed closure time in the radiological analysis in NUREG-1793, Section 6.2.4.13. The 30 second closure times of the two new 15.2-cm (6-in) vacuum relief valves would be bounded by the current design, as the radiological consequences following a LOCA have already been found acceptable for two open 40.6-cm (16-in) valves, closing in 30 seconds.

To analyze the LOCA containment minimum backpressure, containment purge was assumed to be in operation at time zero and air was vented through both the 40.6-cm (16-in) diameter containment purge supply and exhaust lines until the isolation valves fully closed. These valves were modeled to close 12 seconds after the 55.2 kPa (8 psig) closure setpoint was reached, as described in DCD Tier 2, Section 6.2.1.5.3. This closure time was found acceptable as an assumed closure time in the ECCS analysis (reflood backpressure) in NUREG-1793, Section 6.2.4.13. In DCD, Tier 2, Section 9.4.7.2.1, the maximum time for closure of the vacuum relief valves, 30 seconds, was evaluated for its impact on the calculation of the LOCA backpressure. The minimum containment backpressure following two, 15.24-cm (6-in) valves closing in 30 seconds, is expected to be bounded by the containment backpressure resulting from two, 40.64-cm (16-in) valves closing in 12 seconds.

In NUREG-1793, Section 6.2.6, the staff reviewed the applicant's proposed containment leakage rate testing program for AP1000 facilities described in DCD Tier 2, Section 6.2.5 and in the proposed TS of DCD Tier 2, Chapter 16. The staff reviewed the information in the DCD for conformance to 10 CFR Part 50, Appendix J, and to GDC 52 and GDC 53. GDC 52 requires the containment and associated equipment to be designed such that the periodic containment integrated leakage rate tests can be conducted at containment design pressure. GDC 53 requires that the containment allow periodic inspection, surveillance, and testing of certain SSCs.

The staff used the guidance, staff positions, and acceptance criteria of NUREG-0800 Section 6.2.6 and RG 1.163, "Performance-Based Containment Leak-Test Program," in conducting its review.

The staff concluded that the AP1000 containment leakage rate testing program complied with the acceptance criteria of Section 6.2.6 of NUREG-0800 by satisfying the containment leakage rate testing requirements of GDC 52 and GDC 53, and Appendix J to 10 CFR Part 50.

The proposed vacuum relief system valves, VFS-PL-V800A/B and VFS-PL-V803A/B are also part of the containment isolation design and have been added to AP1000 DCD Tier 2, Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves." These new containment isolation valves will be included in the AP1000 containment leak rate test program, and will be Type C tested, as indicated in DCD Tier 2, Table 6.2.3-1. Adding the new

containment isolation valves to the already approved containment leak rate test program, meets the acceptance criteria in Section 6.2.6 of NUREG-0800 for containment leak rate testing of the new valves.

With regard to 10 CFR 52.47(b)(1), the applicant has added appropriate ITAAC to Tier 1. The applicant has included ITAAC to verify the valves function in both the open and closed directions. The staff finds the additional ITAAC acceptable.

23.W.3.3 Valve Design, Qualification, and Testing

The staff reviewed the functional design, qualification, and IST program descriptions for the valves to be used in the applicant's proposed containment vacuum relief system. The staff provided comments to the applicant related to the functional design, qualification, and IST program descriptions for the valves to be used in the proposed containment vacuum relief system. The applicant provided responses to all NRC requests for information. The staff also performed audits of the valve design specifications and data sheets referenced in the proposed design changes.

The proposed AP1000 containment vacuum relief system design includes parallel motor-operated butterfly valves VFS-PL-V800A/B outside containment connected by a single pipe line to parallel check valves VFS-PL-V803A/B inside containment. In the event of a vacuum condition within the AP1000 containment, butterfly valves VFS-PL-V800A/B will receive an automatic electric signal to their battery-powered motor actuators to open the valves. As a result, outside air will flow through the connected piping up to check valves VFS-PL-V803A/B that will open upon a preset differential air pressure across the valves and allow air to enter the AP1000 containment to relieve the vacuum condition. During its review of the applicant's submittal, the staff requested that the applicant clarify the design, qualification, and testing requirements for butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. The applicant provided this additional information in letters dated September 29, 2010, and October 15, 2010.

In Section 3.9.6 of this report, the staff concluded that the AP1000 DCD continues to provide sufficient information to satisfy 10 CFR Part 50 and 10 CFR Part 52, "License, certifications, and approvals for nuclear power plants," for the design aspects of the functional design, qualification, and IST programs for safety-related valves to be used in the AP1000 reactor. The staff noted in Section 3.9.6 of this report that the operational program aspects regarding the functional design, qualification, and IST programs for safety-related valves will be reviewed as part of the evaluation of a COL application referencing the AP1000 certified design. The modifications to the AP1000 DCD related to the functional design, qualification, and IST programs for valves are addressed in Section 3.9.6.

In response to staff comments, the applicant revised Section 9.1.1, "Mechanical Design Requirements," in Section 9.1, "Outboard Motor Operated Valves VFS-PL-V800A/B," of its submittal to specify that the butterfly valves in the AP1000 containment vacuum relief system will be designed in accordance with AP1000 DCD Tier 2, Section 3.9, "Mechanical Systems and Components." The staff describes its review of the design requirements for safety-related MOVs in Section 3.9.6 of this report. For example, as indicated in the letter dated March 5, 2010, the applicant plans to revise Section 3.9 of the AP1000 DCD Tier 2 to specify that qualification of safety-related valves will be in accordance with ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants." The staff accepted the application of ASME Standard QME-1-2007 in 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," with certain staff positions. Also in response to staff comments, the applicant revised Section 9.1.1 of its submittal to specify the design requirements for capacity coefficient and stroke time for full flow capacity of these butterfly valves. The staff finds the reference to the provisions in AP1000 DCD Tier 2, Section 3.9, with the valve-specific design requirements in Section 9.1.1 of the applicant's submittal, to be acceptable for the functional design and qualification of the butterfly valves to be used in the proposed containment vacuum relief system with respect to the design aspects of the AP1000 DC amendment as discussed in Section 3.9.6 of this report.

During its review, the staff requested that the applicant describe the availability of adequate power supplies for motor-operated butterfly valves VFS-PL-V800A/B to perform their design-basis functions. Section 9.1.2, "Valve Electrical Requirements," of the applicant's submittal specifies the design requirements for butterfly valves VFS-PL-V800A/B to be powered from separate Class 1E dc battery sources. Section 9.1.2 also specifies that butterfly valves VFS-PL-V800A/B will be designed to stroke twice for their design-basis operation. Section 9.1.2 indicates that the electrical calculations using the methodology described in IEEE Standard 485, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," will take into consideration the starting current and stroke time for the MOV operations. Section 8.2, "DC Power Systems," in AP1000 DCD Tier 2 describes the application of IEEE-485 for dc power systems in the AP1000 certified design. In addition, the functional qualification of butterfly valves VFS-PL-V800A/B in accordance with ASME Standard QME-1-2007 as accepted in Revision 3 to RG 1.100 will demonstrate the power capability of these MOVs to perform their design-basis functions. The staff finds that the applicant has described the methodology for providing adequate power availability to butterfly valves VFS-PL-V800A/B in an acceptable manner.

In response to staff comments, the applicant revised Section 9.1.3, "Testing Requirements," of its submittal to specify that butterfly valves VFS-PL-V800A/B will be tested in accordance with AP1000 DCD Tier 2, Section 3.9.6, "Inservice Testing of Pumps and Valves." As discussed above, the staff describes its review of the description of the IST program for safety-related valves in Section 3.9.6 of this report. Therefore, the staff finds the reference in the applicant's submittal to the provisions in AP1000 DCD Tier 2, Section 3.9.6 for the description of the IST program for the butterfly valves to be used in the AP1000 containment vacuum relief system to be acceptable with respect to the design aspects of the AP1000 DC amendment as discussed in Section 3.9.6 of this report. The staff will review the operational aspects for the IST program for safety-related valves as part of the evaluation of a COL application referencing the AP1000 certified design.

Section 9.2.1, "Mechanical Design Requirements," of the applicant's submittal specifies the design requirements for the AP1000 vacuum relief containment isolation check valves. ASME OM Code, Table ISTC-3500-1 states that if a check valve used for pressure relief is capacity certified, then it shall be classified as a pressure or vacuum relief valve. Therefore, check valves VFS-PL-V803A/B are within the scope of ASME Code, Section III, Subsection NC-7000, "Overpressure Protection," as defined in Paragraph NC-7110, and require capacity certification as defined in Paragraph NC-7750. In response to staff comments, The applicant revised Section 9.2.2, "Testing Requirements," of its submittal to specify that VFS-PL-V803A/B are vacuum relief valves that will be designed and qualified in accordance with ASME Code, Section III, Subsection NC-7000. During its review, the staff informed the applicant that the design requirements for check valves VFS-PL-V803A/B should be included in Section 9.2.1 (rather than Section 9.2.2). As a result, the applicant modified Section 9.2.1 to specify:

(1) VFS-PL-V803A/B are vacuum relief valves that will be designed and qualified in accordance with ASME Code, Section III, Subsection NC-7000; (2) the valves will be qualified using the provisions in AP1000 DCD Tier 2, Section 3.9; (3) the valves will be designed with an allowable tolerance for the 1.38 kPa (0.2 psi) differential opening pressure; and (4) the valve flow capacity to relieve vacuum conditions to avoid the containment external design pressure from being exceeded. The staff finds the reference in the applicant's submittal to the provisions in AP1000 DCD Tier 2, Section 3.9 and the ASME Code design requirements, with the valve-specific design requirements specified in Section 9.2.1 of the applicant's submittal, to be acceptable for the functional design and qualification of the check valves to be used in the AP1000 containment vacuum relief system.

Section 9.2.2 of the applicant's submittal specifies the testing requirements for AP1000 vacuum relief containment isolation check valves VFS-PL-V803A/B. Because these check valves provide a vacuum relief function with a design opening pressure, the NRC informed the applicant that its submittal should specify that these valves will be tested in accordance with ASME OM Code, Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants." In response, the applicant revised Section 9.2.2 to specify that these check valves are tested in accordance with AP1000 DCD Tier 2, Section 3.9.6 and ASME OM Code, Appendix I. Section 9.2.2 also states that these check valves will be tested in both directions in light of their safety functions in both the open and close directions. Section 9.2.2 specifies that 1.38kPa (0.2 psid) will be used as an acceptance criterion for the check valve opening test. As discussed above, the staff describes its review of the description of the IST program for safety-related valves in Section 3.9.6 of this report. Therefore, the staff finds the reference in the applicant's submittal to the provisions in AP1000 DCD Tier 2, Section 3.9.6 and ASME OM Code, Appendix I, for the description of the IST program for the check valves to be used in the AP1000 containment vacuum relief system to be acceptable with respect to the design aspects of the AP1000 DC amendment as discussed in Section 3.9.6 of this report. The staff will review the operational aspects regarding the IST program for safety-related valves as part of the evaluation of a COL application referencing the AP1000 certified design.

In response to staff comments, the applicant modified Section 6.0, "Containment Isolation Consideration," of its submittal in the "Position" paragraph for 10 CFR Part 50, Appendix A, GDC 54, to clarify that butterfly valves VFS-PL-V800A/B cannot be tested in the direction of containment leakage. Therefore, these butterfly valves will be tested in the reverse direction of containment leakage. This testing is more conservative because the valves will be installed such that the containment pressure will assist in sealing the valve closed to minimize containment leakage. The applicant's submittal indicates the application of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8, "Containment System Leakage Testing Requirements," for containment valve testing as referenced in the AP1000 DCD. The staff requested that the applicant clarify that the butterfly valves will be installed in an orientation that provides for containment pressure to help seal the valve closed. In its response, the applicant revised Section 6.0 of its submittal to specify that the butterfly valves in the containment vacuum relief system will be installed such that containment pressure will assist in sealing the valve closed. The staff finds that the applicant has clarified the orientation of the butterfly valves in the containment vacuum relief system to support the direction of leakage testing for these valves.

As part of its submittal, the applicant plans to modify AP1000 DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirements," to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. The IST table will specify VFS-PL-V800A/B as motor-operated butterfly valves with safety-related missions to maintain close, transfer close, maintain open, and transfer

open; safety functions as active, containment isolation, safety seat leakage, and remote position; ASME Code Class 2 and IST Category A valves; and IST type and frequency as remote position indication and exercise every 2 years, containment isolation leak test, exercise full stroke quarterly and operability test. The IST table will specify check valves VFS-PL-V803A/B as relief valves with safety-related missions as maintain close, transfer close, and transfer open; safety functions as active, containment isolation, and safety seat leakage; ASME Code Class 2 and IST Category AC valves; and IST type and frequency as containment isolation leak test, exercise full stroke every refueling shutdown, and vacuum relief test every 2 years. During its review, the staff requested that the applicant resolve an apparent inconsistency between the IST table and its Note 39. In a letter dated October 15, 2010, the applicant provided a planned revision to the IST table to clarify that the containment vacuum relief butterfly valves will be exercised quarterly and that the containment vacuum relief check valves will be full stroke exercised every refueling shutdown. The applicant's submittal indicates that Note 39 to the IST table will justify the extended test interval for the check valves because of their location inside containment. The staff finds the planned provisions for IST activities for the butterfly valves and check valves in the containment vacuum relief system to comply with the requirements in 10 CFR 50.55a and the ASME OM Code and, therefore, to be acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff requested that the applicant clarify the provisions for inspection and maintenance of the valves in the AP1000 containment vacuum relief system. In its response, the applicant clarified that the inspection and maintenance of butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B are the same as those for other safety-related valves. These valves are required to satisfy the design, inspection, and testing requirements described in the applicable section of the AP1000 DCD. These valves will be designed for the full lifetime of their service, and will not require inspection following individual actuations. Corrective action will be taken for any identified malfunction. Periodic testing and inspection results will be documented and trended. The plant predictive maintenance program and its associated condition monitoring will include these valves. The staff finds that the applicant has clarified the implementation of plant programs for inspection and maintenance for butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B that complies with the NRC regulations and, therefore, is acceptable.

The applicant's submittal indicates that the scope of AP1000 DCD Tier 1, Section 2.2.1, "Containment System," will be expanded to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. Table 2.2.1-3, "Inspections, Tests, Analyses, and Acceptance Criteria [ITAAC]," for the containment system specifies ITAAC for piping and component design in accordance with ASME Code Section III; piping, components, and welds satisfying ASME Code, Section III requirements for integrity; seismic design-basis capability; Class 1E environmental qualification; valve operating times; and valve functional qualification. In its letter dated October 15, 2010, the applicant provided a planned change to Table 2.2.1-3 to specify the closing time for the butterfly valves in the AP1000 containment vacuum relief system. The applicant also indicated that Table 2.7.6-2 in AP1000 DCD Tier 1, Section 2.7.6, "Containment Air Filtration System," will include new ITAAC for the opening time for the butterfly valves in the AP1000 containment vacuum relief system. The staff finds these ITAAC to be acceptable for the butterfly valves and check valves in the AP1000 containment vacuum relief system. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 9.3, "Valve Design Specifications and Datasheets," of the applicant's submittal indicates that the design specifications and data sheets for butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B were available for review at the Westinghouse office. On September 16, 2010, the staff reviewed the referenced documents at the Westinghouse office. During this audit, the staff found that the referenced valve specifications and data sheets had not been updated to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. As a result, the applicant revised Section 9.3 in its submittal to describe the data sheets to be prepared for the butterfly valves and check valves to be used in the AP1000 containment vacuum relief system to support the applicable valve design specifications. On October 7, 2010, the staff audited the preliminary valve specifications and data sheets at the Westinghouse office. The staff finds that the applicant has provided an acceptable description of the valve design specifications and data sheets to support the design and qualification of the butterfly valves and check valves to be used in the AP1000 containment vacuum relief system.

ASME Code, Section III, Article NE-7000, "Overpressure Protection," Subsection NE-7150, "Acceptable Pressure Relief Devices," specifies in Article NE-7152 that vacuum relief devices shall meet the construction requirements applicable to Class 2 valves. The ASME Code also states that valve devices intended to provide vacuum relief, and which are operated by indirect means depending upon an external energy source, are not acceptable unless the following conditions are met: (1) at least two independent external power-operated valve and control systems are employed so that the required relieving capacity is obtained if any one of the valve systems should fail to operate when called upon to do so; and (2) at least one self-actuating vacuum relief device of equivalent relieving capacity is provided in series with each of the external power-operated valves. The ASME Code indicates that acceptable self-actuating vacuum relief devices include balanced self-actuating, horizontally installed, swing disk valves, with provisions for adjustment for the differential pressure under which the valves will operate.

For the proposed AP1000 containment vacuum relief system, Sections 9.1.1 and 9.2.1 of the applicant's submittal specify that butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B will be designed as ASME Code, Class 2 valves. Motor-operated butterfly valves VFS-PL-V800A/B will receive an automatic electric signal to their battery-powered motor actuators to open the valves from separate Class 1E dc battery sources. Check valves VFS-PL-V803A/B will be designed as swing check valves installed in the horizontal direction with a 1.38 kPa (0.2 psi) differential opening pressure. The design of the proposed AP1000 containment vacuum relief system includes parallel check valves VFS-PL-V803A/B in series with the parallel butterfly valves VFS-PL-V800A/B. The staff finds that the design of the vacuum relief devices for the proposed AP1000 containment vacuum relief system satisfies the provisions in Subsection NE-7150 of ASME Code, Section III.

23.W.3.4 Instrumentation and Control

As documented in Chapter 7 of NUREG-1793, the staff reviewed and approved the engineered safety features and safety-related display information as specified in Sections 7.3 and 7.5, respectively, of the AP1000 DCD, Revision 15. The staff reviewed these proposed design changes using the review procedures described in NUREG-0800 Sections 7.3 and 7.5.

The applicant created a new functional diagram Figure 7.2-1 (Sheet 19 of 21) for controlling the proposed containment vacuum relief MOVs from the PMS. The proposed vacuum relief system MOVs are designed to close within 30 seconds by using the existing VFS isolation signal from the PMS, which includes closure from an automatic containment isolation signal, a High-1 containment radiation signal, a manual containment isolation, or a manual containment cooling

signal. In accordance with these proposed design changes, the normally closed vacuum relief system MOVs would open automatically during either of the following conditions:

- 2-out-of-4 coincidence logic for a Low-2 containment pressure condition
- manual actuation of either of the two momentary controls

The 2-out-of-4 logic also receives signals that indicate whether divisions have been bypassed. If one division is bypassed for test or maintenance, the logic will automatically be modified to 2-out-of-3 coincidence logic. Actuating either of the two momentary manual controls in the MCR would actuate all applicable divisions. Separate momentary controls are also provided in the MCR for manually resetting the containment vacuum relief system. In these proposed design changes, the open control function has priority over the closing control function for the containment vacuum relief MOVs. In addition, in order to prevent the purge line isolation inside valve from being opened simultaneously with the vacuum relief system MOVs, interlock logic is implemented to make sure that the vacuum relief system MOVs could not be opened unless the purge line isolation inside valve is closed; if open, vacuum relief system MOVs would close.

While the vacuum relief system MOVs are open, the air flow would be into the containment, which is at a vacuum. Once the vacuum condition inside containment is reduced to near ambient pressure conditions, the open signal would automatically clear. This would allow the vacuum relief system MOVs to close automatically in the event that a containment isolation signal or high radiation signal is present.

The applicant included the necessary design change information in the revised Tier 2, Section 7.3.1, Tables 7.3-1 and 7.3-3, new functional diagram Figure 7.2-1 (Sheet 19 of 21), and Tier 1, Table 2.5.2-3 in the DCD. The applicant also added a new Section 7.6.2.4 in Tier 2 to address the interlock for the containment vacuum relief isolation system, which was added to Tier 1, Table 2.5.2-7 as well for the PMS Interlocks. Since the above logic included for the proposed containment vacuum relief system is designed with division redundancy and bypass capability, the staff concludes that the proposed containment vacuum relief system meets the applicable criteria of protection, reliability, and testability as required in 10 CFR 50.55a(h), and GDC 20 and GDC 21. The applicant added the status of the containment vacuum relief MOVs to Table 7.5-1 for post-accident monitoring, and revised Tier 1, Tables 2.5.2-3 and 2.5.2-4 to add the automatic and manual control functions in the PMS for the containment vacuum relief MOVs. The applicant revised Tier 1, Table 2.5.2-5 and Tier 2, Table 18.12.2-1 to include the display and control of the containment vacuum relief MOVs as part of the minimum inventory for the AP1000 DCD. On the basis of its review of the above changes, the staff finds that the design of the new containment vacuum relief system proposed by the applicant meets the applicable instrumentation and control requirements as mandated in GDC 13 and GDC 19, and also complies with the regulatory guidance in RG 1.97.

The staff finds that the proposed instrumentation and controls design changes meet the relevant requirements in 10 CFR 50.55a(h); 10 CFR 52.47; and 10 CFR Part 50, Appendix A, GDC 13; GDC 19; GDC 20; and GDC 21. The proposed design changes also conform to the related guidance in RG 1.97. Therefore, the staff concludes that the instrumentation and controls aspects of the proposed containment vacuum relief system are acceptable.

23.W.3.5 Technical Specifications

With respect to the proposed TS requirements in TS 3.3.2 and TS 3.6.10, and the proposed changes to TS 3.6.4 and 3.6.5, the staff finds these additions and changes acceptable because

they conform to guidance in the Westinghouse STS (NUREG-1431, "Standard Technical Specifications — Westinghouse Plants"), and reflect the containment systems design information in DCD Sections 7.3 and 9.4.7, and their associated analyses described in DCD Section 6.2. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.W.4 Conclusion

On the basis of its review and confirmatory calculations, the staff finds that the external pressure analysis meets the acceptance criteria from NUREG-0800 Section 6.2.1.1.A, and thus satisfies GDC 16.

On the basis of its review of the containment isolation design aspect of the proposed vacuum relief design, the staff concludes that the design complies with the acceptance criteria in Section 6.2.4 of NUREG-0800, including 10 CFR 50.34(f)(2)(xiv), and CSB BTP 6-4.

On the basis of its review, the staff concludes that the proposed addition of the vacuum relief valves to the already certified AP1000 containment leakage rate testing program complies with the acceptance criteria of Section 6.2.6 of NUREG-0800. Compliance with NUREG-0800 acceptance criteria provides adequate assurance that containment leaktight integrity can be verified before initial operation and periodically throughout its service life. Compliance with the criteria in Section 6.2.6 of NUREG-0800 constitutes an acceptable basis for satisfying the containment leakage rate testing requirements of GDC 52, GDC 53, and Appendix J to 10 CFR Part 50.

The staff reviewed the functional design, qualification, and IST program description for the valves to be used in the proposed AP1000 containment vacuum relief system in accordance with the regulatory basis described in this safety evaluation. Based on its review, the staff concludes that the applicant has provided an acceptable description of the design aspects of the functional design, qualification, and IST programs for the valves to be used in the AP1000 containment vacuum relief system. The staff will review the operational aspects regarding the functional design, qualification, and IST programs for safety-related valves as part of the evaluation of a COL application referencing the AP1000 certified design.

The staff finds that the proposed instrumentation and controls design changes meet the relevant requirements in 10 CFR 50.55a(h), and 10 CFR Part 50, Appendix A, GDC 13, GDC 19, GDC 20, and GDC 21.

On the basis of its review, the staff finds that the proposed changes to the TS that have been established for the containment systems meet the requirements of 10 CFR 50.36.

In addition to the specific DCD changes discussed above in this SER section, the applicant proposed additional changes to several sections, tables, and figures in AP1000 DCD Tier 2 Sections 3.2, 3.7, 3.8, 3.9, 3.11, 7.2, 7.7, 18.12, 19.55, 19.59, and Appendices 1A, 3I and 9A to incorporate the design, testing, and operation of the new containment vacuum relief system and its individual components. The staff has reviewed those proposed DCD changes and finds them to reflect the design, testing, and operation of the new containment vacuum relief system in an acceptable manner.

23.X Changes to the Passive Containment Cooling System

23.X.1 Description of Proposed Changes

The applicant has proposed changes to modeling of the Passive Containment Cooling System (PCS) to correct an error in the time for PCS to begin steady state film coverage of the containment vessel shell which is determined using a scaling factor from AP600 full scale 1/8 sector testing. The applicant assessed the impact of these changes to the PCS in APP-GW-GLR-096, "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analysis," Revision 2, and submitted the report to the NRC in a letter dated March 10, 2011.

This report provides a scaling calculation of the time for PCS to begin steady state film coverage, a discussion of how these changes impact design basis test results, and results of evaluations of the limiting design basis accidents (DBA). Appendix B of APP-GW-GLR-096 contains proposed DCD changes. The changes that impact Section 6.2.1, "Containment Functional Design" of NUREG-1793 are as follows:

- Corrected value of the time for PCS to begin steady state film coverage.

- Update containment response to reflect results of analyses that use the Corrected value of the time for PCS to begin steady state film coverage.

The proposed changes also include revisions to technical specifications (TS). The applicant proposed to change the calculated peak containment internal pressure for the design basis loss of coolant accident to the updated value in TS Bases B 3.6.4, "Containment Pressure," and TS 5.5.8, "Containment Leakage Rate Testing Program".

23.X.2 Regulatory Basis

The following Commission regulations are related to the evaluation of the PCS:

- GDC 38, "Containment Heat Removal," as it relates to the ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a loss-of-coolant accident (LOCA) and to maintain them at acceptably low levels

- GDC 50, "Containment Design Basis," as it relates to demonstrating sufficient margin in accident analysis

- 10 CFR 52.47(c)(2) and 10 CFR 50.43(e), as they relates to design certification analysis and testing in support of a passive plant design.

The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the PCS design and operating information described in DCD Section 6.2. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Chapter 16, "Technical Specifications – Revision 3,

March 2010." Acceptability was based on conformance with the guidance in by SRP Chapter 16.

23.X.3 Evaluation

As described in APP-GW-GLR-096, Revision 2, the applicant estimated the total time to establish steady state film coverage for the AP1000 containment vessel is estimated to be 400 seconds. The applicant modified the WGOTHIC evaluation model that was used in previous DCD analyses to reflect the updated value of time to establish steady state film coverage. The applicant ran the design basis loss-of-coolant accident (LOCA) and main steam line break (MSLB) event cases using this revised model to determine the impact of the delay in PCS water application on the calculated pressure and temperature responses.

In its March 10, 2011, letter, the applicant states that in addition to the PCS water application delay, containment modeling changes included adding an epoxy top coat on the containment vessel to a wainscot height of 7 feet above the operating floor. The applicant states in the same letter that adding an epoxy top coat was to make the containment model and containment pressure analysis consistent with Section 6.1 of the DCD, which states the design requirement for the epoxy top coat. Applicant's sensitivity calculation has shown that the impact on containment peak pressure from adding an epoxy top coat the containment model was 0.01 psi.

Appendix B to APP-GW-GLR-096, Revision 2, provides DCD markup on tables and figures with calculated results for double-ended cold leg guillotine break LOCA, and full main steam line double-ended rupture (DER) at 30 percent and 101 percent reactor power with failure of main steam isolation valve. The results show that the containment pressure and temperature after a LOCA or MSLB would stay below the containment design pressure and temperature.

Figures 8-7 and 8-8 of APP-GW-GLR-096, Revision 2, show comparisons for a double-ended hot leg guillotine (DEHLG) LOCA containment response. As shown in the Figures 8-7 and 8-8, the increased delay time for the application for PCS flow has little or no impact on the calculated peak containment pressure and temperature for this short-term event because the transient pressure peak occurs before PCS flow is initiated.

During an audit performed at the Westinghouse Office in Rockville, Maryland between February 22 and March 4, 2011, the staff reviewed the applicant's calculation report supporting the results of analyses presented in APP-GW-GLR-096, Revision 2.

Based on its review of APP-GW-GLR-096, Revision 2, the staff determined that after correcting the delay in PCS water application in the containment analysis, the calculated pressure and temperature are below the containment design pressure and temperature, and therefore, acceptable. The audit the staff performed confirmed these conclusions.

The staff determined that the change in the delay in PCS water application in the containment analysis did not affect the staff's conclusions as stated in NUREG-1793 and the design is compliant with GDC 38, GDC 50, 10 CFR 52.47(c)(2), and 10 CFR 50.43(e).

During its review, the staff found the applicant's proposed changes to TS Bases B 3.6.4 and TS 5.5.8 acceptable because they reflect the peak containment pressure following a design-basis accident as given in DCD Tier 2 Section 6.2.

In AP1000 DCD Revision 19, the applicant made changes to the text, which are consistent with the proposed changes in the March 10, 2011 letter.

23.X.4 Conclusion

Based on its review the staff concludes that the design changes described above are acceptable and the design is compliant with GDC 38, GDC 50, 10 CFR 52.47(c)(2), and 10 CFR 50.43(e).

23.Y Changes to WGOTHIC AP1000 Containment Evaluation Model Inputs

23.Y.1 Description of Proposed Changes

The applicant has proposed changes to the WGOTHIC AP1000 containment evaluation model (CEM) inputs to address errors and to make updates. The applicant assessed the impact of these changes to the containment response to DBAs in APP-GW-GLR-096, Revision 3, and submitted the report to the NRC in a letter, dated June 14, 2011. The changes include the following, as stated in Section 7.2 of the report:

- The recalculation of the LOCA mass and energy (M&E) releases
- The removal of the inorganic zinc coatings within the LOCA zone of influence and within the maximum flood elevation inside the containment
- The revision of the specific heat for the inorganic zinc and epoxy coatings
- The revision of the material properties for the containment shell
- The correction of the modeling of heat transfer from the containment vessel shell below the operating deck
- The credit for some existing thermal conductors for platforms/gratings
- The consideration of the accumulator nitrogen gas after injection

Changes to the calculation of LOCA M&E releases are identified in Section 7.3 of APP-GW-GLR-096. They include:

- An increase in the SG pressure at the tube bundle
- An increase in the vessel metal mass
- A reduction in the SG tube heat transfer area
- A reduction in core power
- An increase in the RCS fluid volume
- An increase in the main feedwater flow
- A reduction in the equilibration temperature

Appendix B of APP-GW-GLR-096 contains proposed DCD changes. The changes to the WGOTHIC AP1000 CEM input resulted in a small increase of containment peak containment pressure, which impacts a number of sections of the DCD and Section 6.2.1, "Containment Functional Design" of NUREG-1793. Crediting some existing thermal conductors for

platforms/gratings, which were not credited in the CEM before, also impacts Section 21, "Testing and Computer Code Evaluation" of NUREG-1793.

The proposed changes also include revisions to TS. The applicant proposed to change the calculated peak containment internal pressure for the design basis LOCA to the updated value in TS 5.5.8, "Containment Leakage Rate Testing Program," and TS Bases B 3.6.4, "Containment Pressure."

23.Y.2 Regulatory Basis

The Commission regulation related to the evaluation of the proposed changes to the CEM is GDC 50, "Containment Design Basis." GDC 50 requires the demonstration of sufficient margin in the containment for design basis accident analysis. The staff evaluated the proposed changes using the guidance provided in NUREG-0800 Sections 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments"; and 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)." Acceptability was based on conformance with the guidance in NUREG-0800 Sections 6.2.1.1.A and 6.2.1.3.

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the PCS design and operating information described in DCD Section 6.2. The staff evaluated the proposed changes using the guidance provided by NUREG-0800 Chapter 16, "Technical Specifications." Acceptability was based on conformance with the guidance in NUREG-0800 Chapter 16.

23.Y.3 Evaluation

As described in APP-GW-GLR-096, Revision 3, the applicant modified the WGOTHIC CEM inputs that were used in previous DCD analyses to reflect the corrected and updated values. The applicant ran the design basis LOCA and MSLB event cases using this revised model to determine the impact of the input changes on the calculated pressure and temperature responses.

23.Y.3.1 LOCA Mass and Energy (M&E) Releases

The staff reviewed the long-term LOCA M&E release calculations and the methodology as it applies to the AP1000. The long-term LOCA M&E release calculation provides a key input for the containment evaluation model. The blowdown phase M&E releases were calculated using the SATAN-VI computer code, as described in WCAP-10325-P-A (proprietary) and WCAP-10326-A (nonproprietary), May 1983). The staff has previously determined that the SATAN-VI LOCA blowdown computer program is acceptable for use in obtaining LOCA M&E releases for the LOCA blowdown phase for containment analyses. SATAN-VI has been approved by the staff for this purpose, as discussed in NUREG-0800 Section 6.2.1.4. Applicability to the AP1000 is addressed in Section 6.2.1.3 of NUREG-1793.

The LOCA M&E release input data is documented in Tables 6.2.1.3-9 and 6.2.1.3-10 of the DCD. As described in Section 7.3 of APP-GW-GLR-096, a number of input values were corrected in the blowdown portion of the LOCA M&E release calculation, along with one input value in the long-term release spreadsheet calculation. These input changes produced slightly higher LOCA M&E releases to the containment.

The seven specific SATAN model changes made to the DCD, Revision 18 analysis were reviewed by the staff and audits were performed at the Westinghouse office in Rockville, Maryland between May 23 and June 24, 2011, to determine that the changes were appropriate. Each change is described in detail below.

a. Steam Generator Pressure at the Tube Bundle

The SATAN-VI model requires an input value for the SG secondary side pressure at the tube bundle. However, the input value that was used to calculate the M&E releases that are documented in DCD Revision 18 was based on the SG secondary side outlet pressure. The value used previously does not maximize the initial stored energy in the SGs secondary side. Correction of this value resulted in a higher calculated containment pressure response. The staff confirmed the code input requirements and the referenced design specifications for SG pressure, and agrees with the applicant that this change is appropriate.

b. Vessel Metal Mass

In the M&E calculation for DCD Revision 18, some of the metal mass associated with the reactor vessel core barrel region was not included in the SATAN-VI model. The total mass of the missing metal was approximately five percent of the total reactor vessel metal mass. The lower original input did not maximize the initial stored energy in the RCS metal. Correction of this value resulted in a higher calculated containment pressure response. The staff confirmed the revised input to the SATAN-VI M&E calculation.

c. Steam Generator Tube Heat Transfer Area

The tube heat transfer area input value for each SG in the M&E calculation for DCD Revision 18 was approximately seven percent larger than the correct value. Correction of the tube heat transfer area reduces the heat transfer between the RCS and SGs during the blowdown phase, but does not impact the SG energy release rate that is assumed in the long-term spreadsheet calculation. The staff confirmed the revised input to the SATAN-VI M&E calculation and agrees with the applicant that the long-term release rate assumed is unaffected by the change in input area.

d. Core Power

The power level input for the LOCA M&E release calculation was changed from the DCD Revision 18 analysis value of 3415 MWt to 3400 MWt. The original value included 15 MWt of RCP energy that is generated when all four pumps are energized. The RCPs are tripped on a LOCA. Additionally, the analysis methodology for a design basis large break LOCA assumes a loss of offsite power coincident with LOCA initiation. With a loss of offsite power, the RCPs would trip and begin coasting down. The SATAN code input for power is the licensed core power plus the plant-specific power uncertainty, which is included for conservatism. The input is used to initialize the reactor decay heat calculation. Including the energy that is equivalent to the energy generated by the RCPs when electrically powered is not necessary for the SATAN input because RCP energy does not contribute to the decay heat from the fuel. The revised core power is the licensed core power. The implemented change does not alter the modeling characteristics of the RCPs from the prior LOCA M&E release analysis or the approved methods in WCAP-10325-P-A. Changing this core power input value results in a slight reduction in the decay heat contribution to the event. With a loss of power, the RCPs will not continue to operate. However, the staff was concerned about the treatment in the analysis of

the additional heat contributed by the RCPs upon coastdown. In an audit discussion and subsequently in a letter dated July 7, 2011, the applicant provided further justification for excluding the pump energy from the initial core power. The pump coastdown energy was shown by the applicant to be an extremely small contribution to the total energy release during a LOCA. Independent calculation using design data provided by the applicant confirmed the pump coastdown energy addition and duration. As a result, the staff finds the core power change in SATAN acceptable.

e. Reactor Coolant System Fluid Volume

The LOCA M&E release methodology uses fluid volumes that have been determined from engineering drawings. This is referred to as a "cold" condition, so the fluid volumes that are used as input to the SATAN-VI model were increased to account for thermal expansion and measurement uncertainty. The total fluid volume of the RCS in the original calculation for DCD Revision 18 was slightly underestimated. The applicant has corrected the values in the new M&E calculations and as such the staff finds the change acceptable. Although the change is small and the overall impact is insignificant the correct values are now used.

f. Feedwater Flow Rate

The typical simplifying assumption for large break LOCA M&E releases analyses is that main feedwater flow equals steam flow at full power operation. While performing the transient initialization checks to assure that the correct steam flow was achieved after changing the initial steam pressure in the SGs, the applicant discovered that the value that was used in the previous analysis for the main feedwater flow rate for each SG was about 50 percent too low. Correction of the initial feedwater flow rate input value increases the stored energy in the secondary side of each SG after blowdown. This increased the long term energy release rate and the subsequent calculated containment pressure response. The staff audited the revised calculation, and agreed with the applicant that the effect on the containment pressure results is small. The applicant has corrected the value and the staff finds the revised feedwater flow acceptable.

g. Equilibration Temperature

The long term LOCA M&E release methodology uses a temperature "setpoint" to define the end point for the removal of the stored energy remaining in the primary system metal, the secondary system metal and secondary side system fluid inventory. The equilibration temperature that was used in the previous analysis was higher than the value recommended in Section 14 of WCAP-15846. Reducing the equilibration temperature would increase the long-term LOCA M&E releases and increase the subsequent calculated containment pressure response. During the audits, the staff reviewed the long-term M&E release spreadsheet calculation in which this temperature input is used. The staff confirmed that the appropriate value was used. As such, the staff finds the change acceptable.

23.Y.3.2 Inorganic Zinc Coatings within LOCA Zone of Influence and Maximum Flood Elevation inside Containment

The applicant previously made a design change to reduce the potential post-LOCA amount of debris inside containment to support Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance" evaluations. This design change replaces the inorganic zinc coatings on surfaces inside containment that are within the zones of influence of

a LOCA and within the flood elevation (this does not include the containment vessel) with epoxy. To reflect this change in the CEM input, the applicant replaced the inorganic zinc coating material properties for all of the internal heat sinks in the CEM with those of epoxy paint.

Epoxy has lower volumetric specific heat and thermal conductivity values than inorganic zinc (Table 6.2.1.1-8 of the DCD). The staff finds that replacing the inorganic zinc coating material properties for all of the internal heat sinks in the CEM with those of epoxy paint is conservative and acceptable.

23.Y.3.3 Specific Heat for the Inorganic Zinc and Epoxy Coatings

Table 6.2.1.1-8 of the DCD and Table 13-49 of WCAP-15846 list the thermal properties for the various materials that are used in the CEM. The applicant replaced the specific heat values for the inorganic zinc and epoxy coatings materials with the lowest values provided by the vendors.

The staff finds that using the lowest specific heat values for the inorganic zinc and epoxy coatings materials in the CEM is conservative and acceptable.

23.Y.3.4 Material Properties for the Containment Shell

Section 6.2.1.1.2 of the DCD states that the containment vessel is designed and constructed in accordance with the ASME Code, Section III, Subsection NE, "Metal Containment," as described in Section 3.8.2. The applicant updated the containment vessel material properties (density, specific heat, thermal conductivity) to reflect the values specified by the ASME Code and biased for minimum heat removal in the CEM (Table 13-136 of WCAP-15846).

The staff finds that using the ASME Code-specified material properties are consistent with the design of the containment vessel and biasing the property values is conservative and acceptable.

23.Y.3.5 Heat Transfer from Containment Vessel Shell below Operating Deck

A portion of the containment shell below the operating deck is incorrectly modeled in the previous CEM as an internal heat sink with heat transfer from both inside and outside surfaces. To correct the incorrect modeling, the applicant proposed to model the surface as insulated, transferring no heat from the outside surface.

The staff finds the above model correction conservative because it ignores any heat transfer to the outside through the portion of containment shell of interest and, therefore, acceptable.

23.Y.3.6 Crediting Selected, Existing Thermal Conductors

The applicant has changed the CEM to credit a few of the many existing heat structures in containment that were not credited for heat removal before. This change has the effect of reducing peak containment pressure. Section 1.1 of APP-GW-GLR-096, Revision 3, states the following:

> Some of the existing thermal conductors that are in the containment evaluation model are not credited for heat and mass transfer in the peak pressure/temperature analyses because they were not part of the official design at the time the model was first developed. This includes the platforms and gratings inside containment. As shown in

Section 13 of WCAP-15846 [Reference 2], these thermal conductors were modeled as insulated on both sides to prevent heat and mass transfer to them. As described in Section 7.2, heat and mass transfer to some of these conductors have been "turned on" now in the containment evaluation model for the peak pressure/temperature analyses to help offset the impact of the higher LOCA M&E releases.

Figure 7.2-1 of APP-GW-GLR-096, Revision 3, schematically shows the locations of the existing thermal conductors that are now being credited for heat and mass transfer in the peak pressure/temperature calculations.

Heat transfer to the thermal conductors in the "dead-ended" compartments is allowed only during the initial blowdown phase. The accumulator compartments (PXS-A, PXS-B) are located below the core makeup tank compartment and are only open at the top. This inhibits the circulation of break flow to these compartments and they are described as dead-ended. Table 7.4-2 of APP-GW-GLR-096, Revision 3, lists all the heat sink thermal conductors that are currently credited in the CEM for heat transfer in each dead-ended compartment.

During the calculation audits, the staff requested further justification for crediting heat and mass transfer to the dead-ended compartments below the operating deck during the blowdown phase. In a letter dated July 7, 2011, the applicant provided a discussion of the large scale test (LST) benchmarks and the correlation to the AP1000 containment. In the LST benchmarks, dead-ended compartments were observed to be well-mixed. Although the LST volumes and flow path areas could not be directly correlated to the AP1000 volumes and flow path areas, the staff agrees with the applicant that the results are representative of the actual plant conditions. The staff also examined other experimental containment benchmarks referenced in APP-GW-GLR-096, such as the HDR and Battelle Model Containment tests to confirm that the applicant's treatment of dead-end compartments is appropriate.

Heat transfer to certain thermal conductors in the compartments above the operating deck and in "flow-through" compartments below the operating deck is allowed for the entire transient. SG, CMT, and vertical access tunnel (VAT) compartments, which have multiple flow paths with large areas from adjacent compartments allowing flow of steam through each compartment, are describes as flow-through compartments. Figure 7.2-1 also shows the paths for circulation between the various compartments below the operating deck. The LOCA break location is assumed to be in the lower east SG compartment. Section 7.4.1 of APP-GW-GLR-096, Revision 3, provides the following reasons for assuming SG, CMT, and VAT compartments as flow-through compartments:

- The steam release is low, as the hot leg and cold legs are located at approximately the 100-ft elevation. The rising jet/plume will help circulate the containment atmosphere.

- The PCS is actuated to provide cooling by sensible heating and evaporation of water that is evenly distributed on top of the external surface of the containment vessel. This will induce falling wall plumes that will help circulate the containment atmosphere.

- The containment atmosphere is well mixed and the temperature field is not stratified during, and just after, the large LOCA blowdown occurs.

- The concrete, sump, and RCS metal heat sources are located low in the containment. The heat release from these sources will induce rising thermal plumes to help circulate the containment atmosphere.

- The SG, CMT, and VAT compartments that are located below the operating deck are connected to one another and are not dead-ended. This promotes circulation and mixing between these compartments and the volume above the operating deck.

The applicant conservatively ignored heat transfer to the floors of all the compartments because the formation of a liquid layer will inhibit condensation heat transfer to those heat sinks. The staff's SER for the CEM originally included the restriction that heat structures in dead-ended compartments below the operating deck be turned off after blowdown. There is greater uncertainty in flow paths below the operating deck and this restriction was imposed, in part, to ensure conservatism in the overall calculation. The applicant has demonstrated that there remains conservatism in the overall calculation when crediting a few of the heat structures below the operating deck. As described in Section 7.2 of APP-GW-GLR-096, Revision 3, the applicant conservatively ignored in the CEM a significant heat conductor sinks that are available in the AP1000 containment. These include about 35,000 pounds of carbon steel in the CMT and VAT compartments and close to 300,000 pounds of carbon steel above the operating deck (the jib crane, the integrated head stand, etc.). In light of the significant mass of heat structures above and below the operating deck that are not credited, the staff finds that crediting a few is acceptable. This approval does not imply additional heat structures below the operating deck will be approved by the NRC in the future.

Based on its review of APP-GW-GLR-096, Revision 3, the staff determined that crediting the existing thermal conductors for heat and mass transfer in the CEM is consistent with the guidance provided in NUREG-0800 Section 6.2.1.1.A and acceptable.

23.Y.3.7 Release of Accumulator Nitrogen Gas after Coolant Injection

Energy from the pressurized nitrogen gas inside the accumulators is used to force the water from the accumulators when the RCS pressure drops below 700 psig. The gas will expand into the DVI lines after all of the water from the accumulators has been discharged and collect at the high points in the RCS. Some of the gas will be released into the containment atmosphere. The previous CEM did not account for the contribution of the nitrogen gas to the containment peak pressure in the containment response analyses. The applicant proposed to update the CEM to include the effect of accumulator nitrogen gas released into the containment.

The applicant assumes an isothermal expansion of nitrogen gas from accumulators to the containment, conservatively ignoring heat transfer needed from the containment atmosphere to the nitrogen gas for such an expansion. The nitrogen gas is released directly from the accumulators to the containment atmosphere over a period of 60 seconds after the water discharge ends. Section 7.2 of APP-GW-GLR-096, Revision 3, states that "[t]he model is not sensitive to this assumption since the gas release occurs before the time of peak pressure (typically around 1500 seconds after event initiation)." The staff finds that the containment peak pressure and temperature are not sensitive to the assumed duration of nitrogen release because the peaks occur much later after the accumulators empty for double ended guillotine cold leg break LOCA and much before the accumulators empty for double ended guillotine hot leg break LOCA and MSLB accidents. The staff finds the applicant's proposed modeling of accumulator nitrogen gas released into the containment conservative and, therefore, acceptable.

As a result of making changes to the WGOTHIC AP1000 CEM inputs to address the errors and updates discussed above and the two errors stated in Section 23.X of this report, the calculated containment peak pressure increased from 398.5 kPa (57.8 psig) stated in AP1000 DCD Revision 15 to 402.0 kPa (58.3 psig) (Figure B-2 of APP-GW-GLR-096, Revision 3), which is below the containment design pressure of 406.8 kPa (59 psig). Considering the conservatisms build into the WGOTHIC AP1000 CEM and that the containment peak pressure stays below the containment design value, the staff finds that the updated containment peak pressure of 402.0 kPa (58.3 psig) is acceptable.

During the audits, the staff reviewed the applicant's calculation reports supporting the results of analyses presented in APP-GW-GLR-096, Revision 3.

Based on its review, the staff determined that after making changes to the WGOTHIC AP1000 CEM inputs, the calculated pressure is below the containment design pressure and, therefore, acceptable. The audits the staff performed confirmed these conclusions.

The staff determined that the changes to the WGOTHIC AP1000 CEM inputs did not affect the staff's conclusions as stated in NUREG-1793 and the design is compliant with GDC 50.

During its review, the staff found the applicant's proposed change to TS 5.5.8 is acceptable because it reflects the peak containment pressure following a DBA as given in DCD Tier 2 Section 6.2.

Based on its review, the staff determined that the DCD changes proposed in Appendix B of APP-GW-GLR-096, Revision 3, resulting from changes to the WGOTHIC AP1000 CEM inputs, are acceptable. The staff confirmed that the applicant has made the proposed changes in DCD, Revision 19.

23.Y.4 Conclusion

Based on its review the staff concludes that the design changes described above are acceptable and the design is compliant with GDC 50. During its review, the staff found that applicant's proposed change to TS 5.5.8 is acceptable.

24. CONCLUSION

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed Westinghouse's changes to the AP1000 design documentation (see Section 1.5 of this report). On the basis of the evaluation described in the AP1000 FSER (NUREG-1793, NUREG-1793 Supplement 1) and this report, the NRC staff concludes that the AP1000 design documentation (up to and including Revision 19 to the AP1000 design control document) is acceptable and that Westinghouse's application for design certification meets the requirements of Subpart B, "Standard Design Certifications," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

NRC FORM 335
(12-2010)
NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG 1793 Supplement 2
Volume 2

2. TITLE AND SUBTITLE

Final Safety Evaluation Report Related to Certification of the
AP1000 Standard Plant Design
Supplement 2
Volume 2
Docket No. 52-006

3. DATE REPORT PUBLISHED

MONTH	YEAR
9	2011

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

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6. TYPE OF REPORT

Safety Evaluation

7. PERIOD COVERED *(Inclusive Dates)*

1/18/2008 to 7/26/2011

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of New Reactor Licensing
Office of New Reactors
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report supplements the final safety evaluation report (FSER) for the AP10 00 standard plant design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) as NUREG-1793 in September 2004 , and NUREG-1793 Supplement 1 in December 2005, to document the NRC staff's technical review of the AP1000 desig n. The application for the AP1000 design was submitted on June 28, 2002, by Westinghouse Electric Corporation (Westingho use) in accordance with Subpart B, "Standard Design Certifications," of Title 10 of the Code of Federal Regulation s (10 CFR) Part 52, and Appendix 0, "Standardization of Design: Staff Review of Standard Designs." This supplement documents the NRC staff's review of Westinghouse's changes to the AP1000 design documentation in the design control document (DCD) since the issuance of Supplement 1 of the FSER. On the basis of the evaluation described in the AP100 0 FSER (NUREG-1793, NUREG-1793 Supplement 1) and this report, the NRC staff concludes that the changes to the DCD (up to and including Revision 19 to the AP1000 DCD) are acceptable and that Westinghouse's application for design certi fication meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 st andard plant design.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

10 SFR Part 52, 10CFR52, AP1000, Westinghouse, passive, shield building

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)
unclassified

(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

Federal Recycling Program